Duke Power[SM]

A Duke Energy Company

DPC-NE-2005A, Revision 3

Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology

K.R. Epperson

J.L Abbott

Submitted: September 1992
Approved: February 1995
Appendix C Approved: November 1996, Rev. 1
Appendix D Approved: June 1999, Rev. 2
Appendix E Approved: September 2002, Rev. 3

Duke Power Company
Nuclear Generation Department
Charlotte, North Carolina

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

September 16, 2002

Mr. M. S. Tuckman
Executive Vice President
Nuclear Generation
Duke Energy Corporation
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SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 AND MCGUIRE NUCLEAR STATION, UNITS 1 AND 2 RE: ACCEPTANCE FOR REFERENCING OF THE MODIFIED LICENSING TOPICAL REPORT, DPC-NE-2005P (TAC NOS. MB3105, MB3106, MB3173, AND MB3175)

Dear Mr. Tuckman:

The Nuclear Regulatory Commission staff has completed its review of the revision to the topical report submitted by the Duke Power Company's (DPC) letters dated September 13, 2001, as supplemented by letter dated August 14, 2002, entitled "Appendix E to Topical Report DPC-NE-2005P, Duke Power Thermal-Hydraulic Statistical Core Design Methodology (Proprietary)". The report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the enclosed NRC evaluation. The evaluation defines the basis for acceptance of the report.

The staff does not intend to repeat its review of the matters described in the report and found acceptable when the report is referenced in a license application, except to ensure that the material presented is applicable to the specific plant involved. Staff acceptance applies only to the matters described in the report.

We request that DPC publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions should include an " -A" (designating accepted) following the report identification symbol.

Should NRC criteria or regulations change so that staff conclusions regarding the acceptability of the report are invalidated, DPC will be expected to revise and resubmit their documentation, or to submit justification for continued effective applicability of the topical report without revision of their documentation.

Sincerely,

J. A. Nakoski for

John A. Nakoski, Chief, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413, 50-414, 50-369 and 50-370

Enclosure: Safety Evaluation

cc w/encl: See next page

SER

Revision 3

McGuire Nuclear Station
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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO APPENDIX E TO TOPICAL REPORT DPC-NE-2005P

DUKE POWER THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

MCGUIRE NUCLEAR STATION, UNITS 1 and 2

DUKE ENERGY CORPORATION

DOCKET NOS. 50-413, 50-414, 50-369, AND 50-370

1.0 INTRODUCTION

By letter dated September 13, 2001 (Reference 1), as supplemented by letter dated August 14, 2002 (Reference 2), Duke Power Company (DPC), a subsidiary of Duke Energy Company, submitted for NRC review and approval, an Appendix E, "McGuire/Catawba Plant Specific Data, Advanced Mark-BW Fuel, BWU-Z CHF Correlation," to the report, DPC-NE-2005P, "Duke Power Thermal-Hydraulic Statistical Core Design Methodology" (Reference 3).

The approval of DPC-NE-2005P in the staff's Safety Evaluation Report, as included in reference 3, acknowledged that the statistical core design (SCD) methodology is direct and general enough that it could be applicable to other pressurized-water reactors (PWRs), however, it was approved with the following restrictions:

(1) The VIPRE-01 methodology for thermal-hydraulic analysis must be approved for use with the core model.

(2) All correlations, including the critical heat flux (CHF) correlation, are subject to the conditions in the VIPRE safety evaluation report (Reference 4).

(3) The methodology was approved for use in DPC plants only.

In addition to the above restrictions, DPC is required to justify on a plant-specific basis the uncertainties and distributions used for each application. The selection of state points used for generating the statistical design limit must also be justified to be appropriate on a plant-specific basis.

2.0 EVALUATION

The submittals contain the plant-specific data and statistical departure from nucleate boiling ratio (DNBR) limits for the McGuire and Catawba Nuclear Stations with the Advanced Mark-BW fuel design using the BWU-Z CHF correlation and provide details of the fuel assembly structural and thermal-hydraulic features unique to the Advanced Mark-BW fuel design.

DPC's August 14, 2002, submittal describes the two separate fuel pellet materials that can be used in this fuel design structure. When used with uranium fuel pellets, the fuel assembly is called Advanced Mark-BW. If used with mixed oxide fuel pellets, the fuel assembly is called Mark-BW/MOX1. Since the issues in this report are applicable to these fuel designs, the term Advanced Mark-BW in this report means both the Mark-BW/MOX1 and the Advanced Mark-BW design. DPC states that the Advanced Mark-BW fuel design is an evolutionary improvement of the successful Mark-BW17 fuel assembly design. The only thermal-hydraulic difference between the Mark-BW17 fuel and the Advanced Mark-BW fuel is the addition of three mid-span mixing grids for the Advanced Mark-BW design. Since the thermal-hydraulic features are the same, the only impact the different fuel rod designs could have on the statistical DNBR limit is in the radial and axial nuclear uncertainties of $F^N_{\Delta H}$ and F_Z in Table E-4 of the submittal (Reference 1).

The analysis is for the McGuire and Catawba Plants (four-loop Westinghouse PWR's) with the Advanced Mark-BW fuel. Approved methodologies including the VIPRE-01 thermal-hydraulic computer code (Reference 5) and the McGuire/Catawba eight-channel model (Reference 6) are used in this analysis.

The SCD analysis described in Reference 1 includes: (1) state points which represent the range of conditions to which the statistical DNB analyses limit will be applied; (2) uncertainties that were selected to bound the values calculated for each parameter at McGuire and Catawba; (3) the transition core model which determines the impact of the geometric and hydraulic difference between the resident 17x17 Westinghouse robust fuel assembly fuel and the new Advanced Mark-BW design; and (4) the statistical DNBR design limit for each state point evaluated that was determined based on the 500 and 6,000 case runs.

The staff's concerns with respect to the SCD analysis in the areas of the applicability of the approved methodologies (References 5, 6, 7, and 8) for the Advanced Mark-BW fuel design, the supporting data bases, and the mixed core application, were responded to by DPC's submittal dated August 14, 2002 (Reference 2), and in discussions on the September 13, 2001, submittal held with DPC on August 28, 2002. DPC indicated in those discussions that : (1) the results of the SCD analyses in Table E-5 are used for selection of a conservative DNBR value for McGuire and Catawba if the statepoints are within the range in Table E-6, otherwise, the DNBR values in Table E-2 from non-SCD analyses will be used; (2) the mixed core flow mismatch can be confirmed from the reactor core monitoring system; and (3) the analyses in Tables E-2 and E-5 were performed as a mixed core to reflect the McGuire and Catawba core designs.

Based on the NRC staff's review of Appendix E to topical report DPC-NE-2005P, and the response to the staff's request for additional information (Reference 2), the staff finds Appendix E, "Use of BWU-Z CHF Correlation with the Advanced Mark-BW Fuel Assembly," to be acceptable because of the following:

(1) NRC-approved methodologies (Thermal-Hydraulic SCD, the VIPRE-01 code, the mixed core model, and the BWU-Z CHF correlation) are used.

(2) The larger of the two correlation limits produced by VIPRE-01 or LYNXT will be used for non-SCD analyses. This DNBR value is 1.19, as shown in Table E-2.

(3) The conservative DNBR value of 1.36 from the 6,000 case runs will be used for SCD analyses.

The staff may audit the data bases used to support this application and the mixed core calculation record file as part of the application review for the first plant that uses this methodology.

3.0 CONCLUSION

Based on the above discussions, the staff concludes that the proposed use of the BWU-Z CHF correlation with the Advanced Mark-BW fuel Assembly for McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2, as described in DPC's submittals dated September 13, 2002, and August 14, 2002, is acceptable.

4.0 REFERENCES

1. Letter from K. S. Canady, DPC, to USNRC, "Catawba Nuclear Station Units 1 and 2, Docket Nos. 50-413, 50-414, McGuire Nuclear Station Units 1 and 2, Docket Nos. 50-369, 50-370, Appendix E to Topical Report DPC-NE-2005P, Duke Power Thermal-Hydraulic Statistical Core Design Methodology," September 13, 2001.

2. Letter from K. S. Canady, DPC, to USNRC, "McGuire and Catawba Nuclear Stations, Units 1 and 2, Docket Nos. 50-369, 50-370, 50-413, 50-414, Topical Report DPC-NE-2005P, Thermal-Hydraulic Statistical Core Design Methodology, Revision 3 (Appendix E); Request for Additional Information," August 14, 2002.

3. Letter from M. S. Tuckman, DPC, to USNRC, "Issuance of Approved Version of DPC-NE-2005P (DPC-NE-2005P-A)," dated August 8, 1995.

4. "Safety Evaluation Report on the VIPRE-01 Code, " May 1986, and "Safety Evaluation by the Office of Nuclear Reactor Regulation Relating to VIPRE-01 Mod-02 for PWR and BWR Applications, EPRI-NP-2511-CCMA, Revision 3," October 30, 1993, USNRC.

5. VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM-A, Vol. 1-4, Battelle Pacific Northwest Laboratories, August 1989.

6. DPC-NE-2004P-A, McGuire and Catawba Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, Revision 1, February 1997.

7. BAW-10199P-A, Addendum 2, "Application of the BWU-Z CHF Correlation to the Mark-BW17 Fuel Design with Mid-Span Mixing Grids," March 2002.

8. DPC-NE-2009P-A, "Duke Power Company Westinghouse Fuel Transition Report," December 1999.

Principal Contributor: T. Huang, DSSA/SRXB

Date: September 16,2002

SER, TER, and Correspondence
Revision 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 8, 1999

RECEIVED
JUN 1 5 1999
W. R. McCollum, Jr.

Mr. W. R. McCollum, Jr.
Vice President, Oconee Site
Duke Energy Corporation
P. O. Box 1439
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SUBJECT: OCONEE NUCLEAR STATION UNITS 1, 2 AND 3 RE: TOPICAL REPORT NUMBER DPC-NE-2005P USE OF BWU-Z CRITICAL HEAT FLUX CORRELATION FOR MARK-B11 FUEL (TAC NOS. M98660, M98661, AND M98662)

Dear Mr. McCollum:

By letter dated April 22, 1997, and supplemented by letters dated September 21, 1998, and May 13, 1999, Duke Energy Corporation requested NRC review and approval of the use of the BWU-Z Critical Heat Flux Correlation for Mark-B11 Fuel, which is described in Appendix D to Topical Report DPC-NE-2005P, "Duke Power Company Thermal - Hydraulic Statistical Core Design Methodology." The submittal contains analyses of the Mark-B11 fuel assemblies (which were analyzed using the BWU-Z critical heat flux correlation) and justifications for the specific uncertainties and distributions used in the application, and for the selected statepoints used to generate the statistical design limit.

The NRC staff was assisted in this review by its consultant, Pacific Northwest National Laboratory (PNNL). Based on the information provided and the analysis and recommendations provided by PNNL, we find the proposed Appendix D to DPC-NE-2005P to be acceptable. However, this approval is subject to the conditions described in the Safety Evaluation, which are also the commitments made in your letter dated May 13, 1999.

Sincerely,

David E. LaBarge, Senior Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-269, 50-270 and 50-287

Enclosure: Safety Evaluation

cc w/encl: See next page

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UNITED STATES
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SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

LICENSING TOPICAL REPORT DPC-NE-2005P

DUKE ENERGY CORPORATION

OCONEE NUCLEAR STATION, UNITS 1, 2, AND 3

DOCKET NOS. 50-269, 50-270, AND 50-287

1.0 INTRODUCTION

By letter dated April 22, 1997 (Reference 1), as supplemented September 21, 1998, and May 13, 1999, (References 2 and 3 respectively), Duke Energy Corporation (DEC), licensee for the Oconee Nuclear Station, Units 1, 2, and 3, requested NRC staff review and approval of Appendix D, "Oconee Plant Specific Data, Mark-B11 Fuel, Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design" (Reference 1), to DPC-NE-2005P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology" (Reference 4). The submittal contains analyses of the Mark-B11 fuel assemblies, analyzed using the BWU-Z critical heat flux correlation, and provides the required justifications for the specific uncertainties and distributions used in the application, and for the selected statepoints used to generate the statistical design limit.

The staff was assisted in this review by its consultant, Pacific Northwest National Laboratory (PNNL). The staff's evaluation includes the licensee's submittal (Reference 1), the licensee's response to the staff's request for additional information (RAI) dated September 21, 1998 (Reference 2), and the licensee's clarification dated May 13, 1999 (Reference 3). The staff has adopted the findings recommended in our consultant's attached technical evaluation report.

2.0 EVALUATION

This review considered Appendix D "Oconee Plant Specific Data, Mark-B11 Fuel, Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design" to DPC-NE-2005(P) "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology". The details of the evaluation are provided in the attachment.

This appendix contains plant-specific data for two-loop Babcock and Wilcox pressurized water reactors and specific limits for the Oconee Nuclear Station with Mark-B11 fuel using the BWU-Z

Enclosure

form of the BWU critical heat flux (CHF) correlation. Approved methodologies, including the VIPRE-01 thermal-hydraulic computer code (EPRI NP-2511-CCM-A, Vol. 1-4) and the Oconee eight and nine channel models (DPC-NE-2003P-A), are used in this analysis.

The statistical core design (SCD) analysis includes: (1) statepoints that represent the range of conditions to which the statistical DNB analyses limit will be applied; (2) uncertainties that were selected to bound the values calculated for each parameter at Oconee and have not changed except for the rod power hot channel factor (Fq), core flow measurement, and departure from nuclear boiling ratio (DNBR) correlation; (3) the statistical DNBR design limit for each statepoint evaluated that was determined based on the 500 and 5000 case runs; and (4) the transition core model that determines the impact of the geometric and hydraulic difference between the resident Mark-B10 series fuel and the new Mark-B11 design. The staff's concerns with respect to the statistical core design analysis in the areas of the applicable range of conditions, the uncertainties for core flow, the hot channel factor Fq and DNBR correlation, and the mixed core penalty were clarified in the licensee's response to the staff RAI (Reference 2).

Based on our review of Appendix D to Topical Report DPC-NE-2005P and the response to the staff's RAI (Reference 2), the staff finds Appendix D, "Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design" to be acceptable. However, this approval is subject to the following conditions that were committed to by DEC in Reference 3:

(1) Omission of the parameter "Fq" from the SCD analysis of the Oconee plant with a new fuel design must be justified for each particular case. Acceptance of its omission in the case of Mark-B11 fuel does not constitute a general approval of its removal from the parameters to be considered in this methodology.

(2) The applicability of a CHF correlation to mixed core geometries is an issue that must be examined for each transition to new fuel to determine if the mixed core non-uniformities take the local hot channel conditions outside the range of applicability of CHF correlation.

(3) The SCD analysis shall be reviewed and revised as needed if the Mark-B11 CHF correlation range of applicability is changed.

3.0 CONCLUSION

Based on our review of Appendix D to the topical report DPC-NE-2005P and supplemental information supplied by DEC, the staff concludes that Appendix D, "Application of BWU-Z CHF Correlation to Mark-B11 Mixing Vane Spacer Grid Fuel Design" is acceptable. However, actions should be taken whenever a new fuel design is introduced, as follows:

1. Omission of the parameter "Fq" from the SCD analysis of the Oconee plant with a new fuel design must be justified for each particular case. Acceptance of its omission in the case of Mark-B11 fuel does not constitute a general approval of its removal from the parameters to be considered in this methodology.

2. The applicability of a CHF correlation to mixed core geometries is an issue that must be examined for each transition to new fuel to determine if the mixed core non-uniformities take the local hot channel conditions outside the range of applicability of CHF correlation.

3. The SCD analysis shall be reviewed and revised as needed if the Mark-B11 CHF correlation range of applicability is changed.

Attachment: Technical Report

Principal Contributor: Tai Huang

Date: June 8, 1999

REFERENCES

1. Letter from M. S. Tuckman to USNRC, Oconee Nuclear Station Docket Nos. 50-269, 50-270, and 50-287, Use of BWU-Z Critical Heat Flux Correlation for Mark-B11 Fuel, April 22, 1997 (Proprietary and Non-Proprietary Information Available).

2. Letter from M. S. Tuckman to USNRC, Response to NRC Request for Additional Information on Appendix D to Topical Report DPC-NE-2005-P, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," September 21, 1998 (Proprietary and Non-Proprietary Information Available).

3. Letter from M. S. Tuckman to USNRC, Oconee Nuclear Station Docket Nos. 50-269, 50-270, and 50-287, Duke Commitment to Conditions of SER and Clarification of Topical Report DPC-NE-2005 Revision Level, May 13, 1999.

4. DPC-NE-2005P-A, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, February 1995.

5. BAW-10199P-A, The BWU Critical Heat Flux Correlations, August 1996.

TECHNICAL EVALUATION REPORT for

DUKE POWER COMPANY THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY; APPENDIX D: OCONEE PLANT SPECIFIC DATA

(DPC-NE-2005(P), Appendix D)

Judith M. Cuta

April 1999

Prepared for
Office of Nuclear Reactor Regulation
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Washington, D.C. 20555
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NRC FIN I2009

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Attachment

SUMMARY

With the corrections to Table D-2 and D-4 provided in the DPC response to the RAI (see Reference 1), the plant specific data for Oconee presented in Appendix D of DPC-NE-2005P are appropriate for use in the SCD analysis. The parameters are for cores containing Mark-B11 fuel, and transition cores containing both Mark-B10 and Mark-B11 fuel assemblies.

BACKGROUND

The Duke Power Company (DPC) statistical core design (SCD) methodology as documented in Topical Report DPC-NE-2005P-A was granted approval by the Nuclear Regulatory Commission on February 24, 1995. This approval acknowledged that the statistical core design methodology is direct and general enough to be widely applicable to any pressurized-water reactor (PWR), with the following restrictions:

(1) The VIPRE-01 methodology for thermal-hydraulic analysis must be approved for use with the core model.

(2) All correlations, including the critical heat flux (CHF) correlation, are subject to the conditions in the VIPRE safety evaluation report (See Reference 2).

(3) The methodology is approved for use in Duke Power Company plants only.

In addition to the above restrictions, DPC is required to justify on a plant-specific basis the uncertainties and distributions used for each application. The selection of statepoints used for generating the statistical design limit must also be justified to be appropriate, on a plant specific basis.

The Topical Report DPC-NE-2005P-A includes an Appendix A with plant specific data for the Oconee plant with Mark-B10 fuel assemblies (B&W fuel), using the BWC critical heat flux correlation to determine the MDNBR limit. The current submittal, Appendix D, is for Mark-B11 fuel assemblies, analyzed using the BWU-Z critical heat flux correlation. The purpose of Appendix D is to provide the required justifications for the specific uncertainties and distributions used in the application, and for the selected statepoints used to generate the statistical design limit.

EVALUATIONS

The presentation of the plant-specific data for Oconee with Mark-B11 fuel is exceedingly terse in Appendix D. This made it extremely difficult to evaluate the justification for any changes in the plant specific data, or the selection of the set of statepoints used in the analysis. In response to a request for additional information (see Reference 1), DPC provided additional documentation of the means of justifying the specific uncertainties for the Mark B-11 core.

There are four major changes in the plant specific parameters used in the analysis of the Oconee core with Mark-B11 fuel (compare Table D-4 of the submittal with Table A-2 of DPC-NE-2005P-A, Appendix A). The core flow uncertainty is larger, the Fq parameter is also larger, the hot channel factor area uncertainty is unchanged (despite significant changes in the geometry), and the parameter Fq" has been omitted entirely from the analysis. These changes were merely reported in the original submittal, and no justification was given. However, additional information supplied in response to the RAI provided adequate justification for the changes. The change in the flow uncertainty is the result of re-calculating the Chapter 15 transients for Oconee with Mark-B11 fuel using the BWU-Z correlation for combinations of 4, 3, and 2 pump operation. The parameter Fq is based on the rod power hot channel factor supplied by the fuel vendor and the approved value of the radial peak uncertainty. The hot channel factor area uncertainty is based on information supplied by the fuel vendor and will be verified by inspection of the final fuel assemblies and components. The specific value of -3.00% for Oconee bounds particular acceptance criteria for the fuel, and cannot be exceeded without invoking additional analyses to determine the effect on the statistical design limit (see Table 7 of DPC-NE-2005P-A).

Omitting the parameter Fq" from the analysis was justified by DPC based on work by other fuel vendors (specifically, in WCAP-8202 and CENPD-207) showing that local heat flux spikes as great as 20% above the local nominal heat flux do not have any noticeable effect on the minimum DNBR. DPC believes that this effect is generic to PWR fuel, and states that it was confirmed to be applicable to Mark-B11 fuel by the fuel vendor. In addition, the parameter Fq" calculated by the vendor is much smaller for this fuel than for Mark-B10 fuel (a value of 1.41% for Mark-B11, compared to 2.08% for Mark-B10.)

The additional information supplied by DPC shows that it is justifiable to omit the parameter Fq" from the SCD analysis of the Oconee plant with Mark-B11 fuel. However, the assertion that the parameter can be omitted in analysis of PWR fuel in general is too broad. Fuel designs developed in future might conceivably have a different sensitivity to this parameter, and DPC should be required to evaluate its applicability to each new fuel design. If it can be omitted for a particular fuel design, DPC must provide justification for such omission, as required by the SER for DPC-NE-2005P-A.

The discussion in Appendix D of the treatment of transition fuel cycles, when Mark-B10 and Mark-B11 fuel would be co-resident in the core, is extremely vague and incomplete. In response to the RAI, however, DPC provided additional details to clarify the method of determining the transition core penalty and implementation of the options for its application. It appears that the methodology used will capture the largest penalty applicable to a specific core design.

Appendix D contains no discussion of the applicability of the BWU-Z CHF correlation to Mark-B11 fuel in mixed cores. This is a serious oversight, since there are marked local pressure drop differences between the Mark-B10 and Mark-B11 fuel assembly designs, even though the overall pressure drop is essentially the same. The local differences (due to differences in the grid design) will result in subchannel flow distributions in the vicinity of the spacer grids that are significantly different from the distributions in a uniform core of Mark-B11 fuel only. Since the BWU-Z CHF correlation is based on data for Mark-B11 fuel only, it is not obvious that the correlation is applicable to cores containing both B-10 and B-11 fuel.

Additional information supplied by DPC referenced CHF testing by the fuel vendor[1] with a 5x5 test assembly simulating mixed core conditions (BAW-10143P-A). For the conditions tested, there was no significant change in the accuracy of the BWC correlation for the bundle modeling a mixed core, compared to results obtained for bundles modeling a uniform core.

[1]Framatome Cogema Fuels, formerly Babcock & Wilcox Fuel Company.

This is evidence that the mixed core conditions do not result in local conditions in the subchannel that are outside the range of the CHF correlation. Thermal-hydraulic calculations with the VIPRE code show that the geometry corresponding to a mixed core of Mark-B10 and Mark-B11 fuel produces local velocity distributions that differ from a uniform core by only about one-fifth as much as the most severe conditions encountered in test data reported in BAW-10143P-A. Based on these factors, DPC concludes that the BWU-Z correlation is also applicable to mixed cores.

This argument has several weaknesses. It is based on data for Mark-B10 fuel, not Mark-B11, and the correlation used to evaluate the data was the BWC correlation, not the BWU-Z correlation. Because CHF correlations are *ad hoc* fits to data sets rather than models based on the physical behavior of the system, there is no reason to suppose as a general rule that what is true of one fuel design and CHF correlation will be true of another fuel design and its CHF correlation. In this particular case, however, it can be argued that there are two main reasons to expect the BWU-Z correlation to behave in essentially the same manner as the BWC correlation for a mixed core of B-10 and B-11 fuel. First, the fuel designs are from the same vendor, and have similar physical geometry. Second, the CHF correlations share a common developmental path, have similar form, and show similar fit to their respective databases. In addition, DPC reports that thermal-hydraulic calculations show the non-uniformities for mixed B-10/B-11 cores will in general be much smaller than the conditions tested using the BWC correlation in the bundle modeling a mixed core for Mark-B10 fuel.

For this case, DPC has shown that the BWU-Z CHF correlation can be expected to be applicable to mixed cores of Mark-B10 and Mark-B11 fuel. However, this conclusion should not be interpreted as laying to rest the generic issue of the applicability of CHF correlations to mixed core geometries. This issue must be examined for each transition to new fuel, to determine if the mixed core non-uniformities result in local hot channel conditions outside the range of applicability of the CHF correlation. At a minimum, subchannel thermal-hydraulic calculations are needed to determine the magnitude of the most severe local velocity depression in the hot channel. Test data obtained in bundles modeling a mixed core may be necessary in some cases to fully resolve the issue.

The description of the range of applicability of the BWU-Z CHF correlation for system pressure was not presented appropriately in the original submittal. Additional information supplied by DPC corrected this deficiency, and a revised version of Table D-2 is included in the response to the RAI (see Reference 1). This table shows that the design limit DNBR of 1.199 is applicable to conditions between 700 and 1000 psia. Below 700 psia, the design limit DNBR is 1.59. In addition, the response states that if a statepoint with pressure below 1000 psia is encountered in an SCD analysis for Oconee, the applicable design limit DNBR will be used and the impact of the higher correlation standard deviation on the statistical design limit will be calculated. If the SDL for the new statepoint is greater than the licensing limit, the higher SDL will be used when analyzing the lower pressure conditions. This procedure is in accordance with the approved methodology, as described in DPC-NE-2005P-A

RECOMMENDATIONS

The plant specific data for Oconee with Mark-B11 fuel and for transition cores containing Mark-B10 and Mark-B11 fuel is appropriate for use in the SCD analysis, based on the justifications provided in the DPC response to the RAI (see Reference 1). This includes the corrections to Table D-2 and D-4 provided in the DPC response to the RAI. However, approval of these parameters for Oconee with Mark-B11 fuel does not constitute generic approval of all matters in Appendix D pertaining to the SCD analysis. Specifically,

- Omission of the parameter Fq" from the SCD analysis of the Oconee plant with a new fuel design must be justified for each particular case. Acceptance of its omission in the case of Mark-B11 fuel does not constitute a general approval of its removal from the parameters to be considered in this methodology.

- The applicability of a particular CHF correlation to mixed core geometries is an issue that must be examined for each transition to new fuel, to determine if the mixed core non-uniformities take the local hot channel conditions outside the range of applicability of the CHF correlation.

The methodology requires that the approved CHF correlation for a given fuel design must be used in the SCD analysis for the Oconee plant. As of this writing, the proposed CHF correlation for Mark B-11 fuel (the BWU-Z correlation, submitted as Appendix E in Addendum 1 of BAW-10199P) is under review and has not yet been approved by the NRC. Any changes to the CHF correlation or restrictions in its application as a result of the NRC review must be evaluated for effects on the application of the correlation to Mark-B11 fuel. If the CHF correlation range of applicability is changed, the SCD analysis must be revised in accordance with the modification, and the correlation must not be used outside the parameter range specified in the safety evaluation report (SER) for application to Mark-B11 fuel.

Duke Energy

M. S. Tuckman
Executive Vice President
Nuclear Generation

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May 13, 1999

U. S. Nuclear Regulatory Commission
Washington D. C. 20555-0001

ATTENTION: Document Control Desk

Subject: Oconee Nuclear Station
Docket Numbers 50-269, -270, and -287
Duke Commitment to Conditions of SER and
Clarification of Topical Report DPC-NE-2005
Revision Level

Duke Energy Corporation Topical Report DPC-NE-2005, "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," was submitted to NRC in September 1992; approval was granted in February 1995. This initial revision included Appendices A and B, which contained Oconee and McGuire/Catawba plant specific data. Subsequent to Rev 0, Duke submitted Appendix C on April 26, 1996 requesting approval for applying the BWU-Z CHF correlation for analyses of the McGuire and Catawba reactor cores with MkBW fuel. Appendix C contained McGuire/Catawba plant specific data for MkBW fuel using the new CHF correlation, BWU-Z. Appendix C was approved on November 7, 1996. Duke placed the November 7, 1996 NRC Safety Evaluation and Appendix C in the back of DPC-NE-2005 and entitled this report DPC-NE-2005P-A, Rev 1. Rev 1 contains no unreviewed technical information. It simply places previously NRC approved documents DPC-NE-2005, Rev. 0 and Appendix C into the same report.

Within this letter, Duke makes the following commitment:

Following NRC's approval of Appendix D (which was submitted in a Duke letter to the NRC dated April 22, 1997) Duke will incorporate the NRC's Safety Evaluation and Appendix D into DPC-NE-2005P-A, Rev. 1 and at this time change the revision level to DPC-NE-2005P-A, Rev 2. No other technical changes will be made.

Further, Duke Energy Corporation accepts the following NRC specified conditions applicable to use of the BWU-Z critical heat flux correlation for Mark B11 fuel in the Oconee reactors:

(1) Omission of the parameter "Fq" from the SCD analysis of the Oconee plant with a new fuel design must be justified for each particular case. Acceptance of its omission in the case of Mark-B11 fuel does not constitute a general approval of its removal from the parameters to be considered in this methodology.

(2) The applicability of a CHF correlation to mixed core geometries is an issue that must be examined for each transition to new fuel to determine if the mixed core non-uniformities take the local hot channel conditions outside the range of applicability of CHF correlation.

(3) The SCD analysis should be revised as needed to reflect the modification if Mark-B11 CHF correlation range of applicability is changed.

If there are any questions, or additional information required, please call R. M. Gribble at (704) 382-6160 or K. R. Epperson at (704) 382-6785.

M. S. Tuckman

M. S. Tuckman

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SER

Revision 1

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 7, 1996

RECEIVED
NOV 14 1996
REGULATORY COMPLIANCE

Mr. M. S. Tuckman
Senior Vice President
Nuclear Generation
Duke Power Company
P. O. Box 1006
Charlotte, NC 28201

SUBJECT: SAFETY EVALUATION ON THE USE OF THE BWU-Z CRITICAL HEAT FLUX CORRELATION FOR MCGUIRE NUCLEAR STATION, UNITS 1 AND 2; AND CATAWBA NUCLEAR STATION, UNITS 1 AND 2 (TAC NOS. M95267, M95268 AND M95333, M95334)

Dear Mr. Tuckman:

By letters dated October 13 and December 4, 1995, as supplemented by letters dated April 26 and September 5, 1996, Duke Power Company requested approval for applying the BWU-Z critical heat flux (CHF) correlation for analyses of the McGuire and Catawba reactor cores with Mark-BW 17x17 type fuel. The BWU-Z CHF correlation for the Mark-BW 17x17 type fuel is one of the three applications stated in Babcock and Wilcox Fuel Company's (BWFC's) (now Framatome Cogema Fuels) Topical Report BAW-10199P, "The BWU CHF Correlations." This topical report was reviewed and approved by the NRC by letter dated April 5, 1996.

Based on its review, the staff finds the proposed application of the BWU-Z CHF correlation for the McGuire and Catawba Mark-BW 17x17 type fuel acceptable. Our safety evaluation, which provides the results of the review, is enclosed.

Sincerely,

Herbert N. Berkow

Herbert N. Berkow, Director
Project Directorate II-2
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Office of Nuclear Reactor Regulation

Docket Nos. 50-369, 50-370
50-413, and 50-414

Enclosure: Safety Evaluation

cc w/encl: See next page

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McGuire Nuclear Station
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

DUKE POWER COMPANY

MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-369 AND 50-370

CATAWBA NUCLEAR STATION, UNITS 1 AND 2

DOCKET NOS. 50-413 AND 50-414

1.0 INTRODUCTION

By letters dated October 13, 1995 (Reference 1) and December 4, 1995 (Reference 2), as supplemented by letters dated April 26, 1996 (Reference 3) and September 5, 1996 (Reference 4), Duke Power Company (DPC or the licensee) requested the use of the BWU-Z critical heat flux (CHF) correlation for analyses of the McGuire and Catawba reactor cores, which consist of a full core of Mark-BW 17x17 type fuel assemblies.

2.0 DISCUSSION/EVALUATION

The licensee submitted Appendix C to DPC-NE-2005P-A to support plant-specific applications to the reload analyses for the McGuire and Catawba plants. Specifically, Appendix C contains the plant-specific data and limits with Mark-BW 17x17 type fuel using the BWU-Z form of the BWU CHF correlation, the VIPRE-01 thermal-hydraulic computer code (Reference 6), and Duke Power Company thermal-hydraulic (T-H) statistical core design (SCD) methodology (Reference 7). The licensee stated that the BWU-Z form of the BWU correlation used in the analyses for the McGuire and Catawba units is exactly the same as the correlation used in BAW-10199P (Reference 5).

In addition, the licensee used the approved method as described in Reference 7 regarding the statepoint propagation. In its calculation of the statistical limit, the licensee increased the number of cases from 3,000 to 5,000 per statepoint. The licensee stated that increasing the number of cases provided higher confidence of defining the bounding behavior and reducing the multipliers. The 5,000-case number was selected due to a balance between computer resources required for the calculation and the reduction in statistical uncertainty to determine a conservative Statistical Design Limit (SDL).

The maximum statepoint statistical value for departure from nucleate boiling ratio (DNBR) for the 5,000-case propagation is given in Table C-4 of Reference 3. This table also contains the values where case propagation is less than the 5,000-case propagation. The 5,000-case value will be used in analyses with the BWU-Z form of the BWU CHF correlation for Mark-BW 17x17 type fuel at McGuire and Catawba.

The statistical design limit given in Table C-4 is applicable to this analysis only when all statepoint parameters fall within the McGuire/Catawba key parameter ranges given in Table C-5 of Reference 3.

DPC has also used the VIPRE-01 thermal-hydraulic computer code (Reference 6) to calculate the measured-to-predicted (M/P) CHF ratios with respect to mass velocity, pressure, or thermodynamic quality. The results show that the average M/P value and the data standard deviation are within 1% of the values reported in BWU CHF correlation (Reference 5).

A comparison between the BWU-Z ranges of applicability for Mark-BW 17x17 type fuel database given in Table 4-1 of Reference 5 and the parameter ranges provided in Table C-1 of Reference 3 shows a 0.01 difference in design limit DNBR using the LYNX and the VIPRE-01 code (1.19 design limit DNBR resulted from the LYNX code versus 1.18 design limit DNBR resulted from VIPRE-01 code). However, DPC will use the larger of the two non-statistical correlation limits.

The staff reviewed the submitals provided by DPC (Reference 1 through Reference 4), and found that the proposed use of BWU-Z CHF correlation is acceptable for use at the McGuire and Catawba plants. This conclusion is based on core analyses that (1) both plants have a full homogeneous core of Mark-BW 17 x 17 type fuel assemblies for upcoming reloads, (2) NRC-approved methodologies (T-H SCD, VIPRE-01, and BWU-Z CHF) are used, (3) the larger of the two correlation limits (VIPRE-01 or LYNX) will be used for non-SCD analyses, and (4) the conservative result from the 5,000-case propagation will be used for SCD analyses.

3.0 CONCLUSION

Based on the above discussions, the staff concludes that the proposed use of BWU-Z critical heat flux correlation for McGuire Nuclear Station, Units 1 and 2, and Catawba Nuclear Station, Units 1 and 2, acceptable.

4.0 REFERENCES

1. Letter from M. S. Tuckman to USNRC requesting review the use of the BWU-Z critical heat flux correlation, dated October 13, 1995.

2. Letter from M. S. Tuckman to USNRC discussing Duke Power Company intent to use of the BWU-Z critical heat flux correlation, dated December 4, 1995.

3. Letter from M. S. Tuckman to USNRC submitting the Appendix to DPC-NE-2005P-A, "McGuire/Catawba Plant Specific Data, Mark-BW Fuel BWU-Z Critical Heat Flux Correlation," dated April 26, 1996.

4. Letter from M. S. Tuckman to USNRC responding to the USNRC's Request for Additional Information regarding Appendix C to DPC-NE-2005P-A, dated September 5, 1996.

5. BAW-10199P, The BWU Critical Heat Flux Correlations, BWFC, November 1994 (Approved by letter from R. C. Jones to J. H. Taylor, dated April 5, 1996).

6. DPC-NE-2004P-A, Duke Power Company McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, December 1991.

7. DPC-NE-2005P-A, Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology, February 1995.

Principal Contributor: T. Huang

Date: November 7, 1996

SER and TER

Revision 0

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

February 24, 1995

Mr. H. B. Tucker
Senior Vice President
Duke Power Company
P.O. Box 1006
Charlotte, NC 28201-1006

Dear Mr. Tucker:

SUBJECT: ACCEPTANCE FOR REFERENCING OF THE MODIFIED LICENSING TOPICAL REPORT, DPC-NE-2005P, "THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY" (TAC NO. M85181)

The staff has completed its review of the subject topical report submitted by the Duke Power Company (DPC) by letter dated September 28, 1992. The report is acceptable for referencing in license applications to the extent specified and under the limitations delineated in the report and the associated NRC evaluation, which is enclosed. The evaluation defines the basis for acceptance of the report.

The staff does not intend to repeat its review of the matters described in the report and found acceptable when the report is referenced in a license application, except to ensure that the material presented is applicable to the specific plant involved. Staff acceptance applies only to the matters described in the report.

In accordance with procedures established in NUREG-0390, DPC must publish accepted versions of this report, proprietary and non-proprietary, within three months of receipt of this letter. The accepted versions shall incorporate this letter and the enclosed evaluation between the title page and the abstract. The accepted versions shall include an -A (designating accepted) following the report identification symbol.

Should NRC criteria or regulations change so that staff conclusions regarding the acceptability of the report are invalidated, DPC will be expected to revise and resubmit their documentation, or to submit justification for continued effective applicability of the topical report without revision of their documentation.

Sincerely,

Robert Jones for

Gary M. Holahan, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

Enclosure: NRC Evaluation

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATING TO THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY

TOPICAL REPORT DPC-NE-2005P

FOR

THE DUKE POWER COMPANY

1.0 INTRODUCTION

By letter dated September 28, 1992 (Ref. 1), the Duke Power Company (DPC) submitted for staff review and approval a report for use in core thermal-hydraulic analysis. DPC submitted additional information on September 29, 1993 (Ref. 2) and again on February 15, 1994 (Ref. 3). This topical report and the supplemental submittals document the development of core thermal-hydraulic analysis based upon the statistical core design (SCD) methodology using the VIPRE-01 computer code (Ref.4) for the DPC plants: McGuire, Catawba, and Oconee nuclear stations.

The SCD method is a thermal-hydraulic analysis technique which computes departure from nucleate boiling (DNB) margin by statistically combining core and fuel bundle uncertainties. The submittal provides a description and justification for applying uncertainties to the DPC DNB ratio (DNBR) limit calculations using a statistical rather than a deterministic (traditional) method. In 1991, NRC, as part of a reload review, approved limited application of the SCD methodology described in References 5 and 6 for McGuire and Catawba applications. By submitting this topical report, DPC proposes to extend the use of this methodology to all DPC plants for the DNB analysis.

The objective of the subject topical report, therefore, is twofold: (1) to formally present a description of the DPC SCD methodology and (2) to justify its use for all DPC plants. In addition, DPC presented its rationale for setting two separate statistical design limits. The underlying core thermal-hydraulic methodology based upon the use of VIPRE-01 was approved for all DPC plants (Refs. 5 and 7).

The SCD methodology presented in the topical report is discussed in generic terms in this review. Where applicable, plant-specific features are discussed separately.

2.0 STAFF EVALUATION

The review of "Thermal/Hydraulic Statistical Core Design Methodology," report DPC-NE-2005P, was performed with technical assistance from International Technical Services (ITS). The ITS review findings are contained in the Technical Evaluation Report (TER) which is attached to this safety evaluation report. The staff has reviewed the TER and has concurred with all its findings.

The traditional method for accounting for design and modeling uncertainties that enter into the determination of a DNBR assumes that key input parameters to the core thermal-hydraulic code are simultaneously at their worst level of uncertainty. The currently licensed SCD methodology for McGuire and Catawba assumes that, while the input parameters are occasionally at their worst case values, the input uncertainties are independent and it is highly unlikely that all the input parameters will take on their worst-case values simultaneously. Therefore, the application of the SCD method differs from the deterministic techniques in that the DNBR limit is obtained from statistical analysis of a series of computations as a result of propagation of uncertainties about a statepoint and associated distribution of the DNBR values. DPC has applied the SCD method to simulate the direct computation of DNBR with VIPRE-01.

The TER discusses the DPC VIPRE methodology, the current and revised SDC methodology, the selection of key parameters and uncertainties, propagation of uncertainties, calculation of the statistical design limit (SDL) for DNBR, flexibility of the methodology, and the statistical DNB behavior and use of two DNBR limits.

During the review of DPC-NE-2005P, questions were raised on the use of two DNBR limits. However, the responses were not detailed enough to supply the information needed for resolving the possible use of two DNBR limits. Therefore, the use of the SCD methodology is approved now for only the single, most-conservative DNBR limit.

3.0 CONCLUSION

The staff has reviewed the subject topical report together with the DPC responses and has found them to be acceptable with respect to documentation of the statistical core design methodology using the VIPRE computer code subject to the following restrictions:

1. The statistical core design (SCD) methodology developed by DPC, as described in the submittal, is direct and general enough to be widely applicable to any pressurized-water reactor (PWR) fuel or reactor, provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the critical heat flux (CHF) correlation subject to the conditions in the VIPRE safety evaluation report (SER). DPC committed in their topical report that its use of specific uncertainties and distributions will be justified on a plant specific basis, and also that its selection of statepoints used for generating the statistical design limit will be justified to be appropriate. This methodology is approved only for use in DPC plants.

2. Of the two DNBR limits, only the use of the single, most-conservative DNBR limit is approved.

4.0 REFERENCES

1. Letter from H. B. Tucker (DPC) to USNRC, submitting "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," DPC-NE-2005P, September 28, 1992.

2. Letter from H. B. Tucker (DPC) to USNRC, submitting "Thermal/Hydraulic Statistical Core Design Methodology, DPC-NE-2005," September 29, 1993.

3. Letter from H. B. Tucker (DPC) to USNRC, submitting "Thermal/Hydraulic Statistical Core Design Methodology, DPC-NE-2005," February 19, 1994.

4. Electric Power Research Institute, "Acceptance for Referencing of Licensing Topical Report VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM Vols. 1-4," May 1, 1986.

5. Duke Power Company, "McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2004, December 1988.

6. Letter from H. B. Tucker (DPC) to USNRC, submitting "Handouts Presented in the October 7 & 8, 1991 Meeting with NRC Staff and Contract Reviewers," October 16, 1991.

7. "Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2003P-A, August 1988.

ITS/NRC/94-3

TECHNICAL EVALUATION: THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY DPC-NE-2005P FOR DUKE POWER COMPANY

P.B. Abramson
H. Komoriya

Prepared for
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555
Under NRC Contract No. NRC-03-90-027
FIN No. L1318

International Technical Services, Inc.
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New York, NY 10170

TECHNICAL EVALUATION:
THERMAL-HYDRAULIC STATISTICAL CORE DESIGN METHODOLOGY
TOPICAL REPORT DPC-NE-2005P
FOR
THE DUKE POWER COMPANY

1.0 INTRODUCTION

DPC-NE-2005P, dated September 1992 (Ref. 1) was submitted by Duke Power Company (DPC) for NRC review and approval. Additional information was submitted on September 29, 1993 (Ref. 2) and on February 19, 1994 (Ref. 3). This topical report and the supplemental submittals document the development of core thermal-hydraulic analysis based upon statistical core design (SCD) methodology using the VIPRE-01 computer code (Ref. 4) for the DPC plants; McGuire, Catawba (M/C) and Oconee Nuclear Stations.

The SCD method is a thermal-hydraulic analysis technique which computes DNB margin by statistically combining core and fuel bundle uncertainties. The submittal provides a description and justification for applying uncertainties to the DPC DNBR limit calculations using a statistical rather than a deterministic (traditional) method. In 1991 NRC, as part of a reload review, approved limited application of the SCD methodology described in References 5 and 6 for M/C applications. By submitting this topical report DPC proposes to extend the use of this methodology for the DNB analysis of all DPC plants.

The objective of the subject topical report, therefore, is twofold: (i) to formally present a description of the DPC SCD methodology; and (ii) to justify its use for all DPC plants. In addition, DPC presented their rationale for setting two separate statistical design limits. The underlying core thermal-hydraulic methodology based upon the use of VIPRE-01 was approved for all DPC plants (Refs. 5 and 7).

The SCD methodology presented in the topical report is discussed in generic terms in this review. Where applicable, plant specific features are discussed separately.

2.0 SUMMARY OF TOPICAL REPORT and SUPPLEMENTS

The topical report DPC-NE-2005 and its associated submittal document descriptions of DPC's VIPRE-01 based statistical core design (SCD) methodology for all of DPC's nuclear stations. The SCD methodology described in the topical has been approved as part of another review in a limited scope. The submittal formalizes the documentation of methodology description and justification for applying uncertainties to the DPC DNBR limit calculations using a statistical rather than a deterministic (traditional) method, since DPC proposes to extend the application of this methodology to

DNB analysis of all DPC plants.

In addition, DPC presented its rationale, and limited justification, for setting two separate statistical design limits due to sensitivity of DNB to the axial power distributions.

3.0 DPC Statistical Core Design Methodology

The traditional method for accounting for design and modeling uncertainties that enter into the determination of a DNBR assumes that key input parameters to the core thermal-hydraulic code are simultaneously at their worst level of uncertainty. The currently licensed SCD methodology for McGuire and Catawba assumes that, while the input parameters are occasionally at their worst case values, the input uncertainties are independent and it is highly unlikely that all the input parameters will take on their worst case values simultaneously. Therefore, the application of the SCD method differs from the deterministic techniques in that the thermal-hydraulic limit analyses are performed by statistical analysis of a series of computations as a result of propagation of uncertainties about a statepoint and associated distribution of the DNBR values. DPC has applied the SCD method to simulate the direct computation of DNBR with VIPRE-01.

3.1 DPC's VIPRE Methodology

DPC has, in place, NRC approved DNB methodology using the VIPRE-01 computer code for all DPC plants. Both the current and revised SCD methodologies are based on use of such NRC approved VIPRE methodology.

3.2 Current Methodology

The current DPC SCD methodology, based upon the B&W SCD method, relies upon the use of the response surface model (RSM) to evaluate the impact of uncertainties associated with each of the key parameters upon the DNB behavior. Therefore, the range of applicability of the SCD method (therefore the RSM) is limited by the range of values from which the composite design points, used to determine the RSM equation, are selected.

In order to overcome the main limitation of the current SCD methodology with respect to the statepoints which fall outside of the SCD range but which must nevertheless be analyzed for certain transients, DPC developed a simplified method which used VIPRE-01 directly and avoided use of the RSM.

The simplified method bypasses the RSM by directly computing DNBR with the VIPRE-01 code based on the values for the key variables generated by the propagation of uncertainties through the use of the Monte Carlo method. An SCD limit is determined for each case as before and compared against the SDL.

3.3 Revised Methodology

The revised methodology, an extension of the simplified method, is similar to other SCD methodologies in that (1) key parameters are selected, (2) their associated uncertainties are propagated about a statepoint and (3) a large number of DNBR's are calculated. However, with this methodology the intermediate step of developing the RSM is eliminated. Instead, statistical behavior at a statepoint is evaluated by observing the distribution of the DNBR values and the mean and standard deviation of DNB for the given conditions computed by use of a Monte Carlo method for selection of values of the independent variable. All DNBR calculations are performed directly by the use of the thermal-hydraulic code VIPRE.

This is an advantage since the applicability issue with the previous method is eliminated. Further, if an assumed uncertainty should become non-bounding, the limiting statepoint can be re-evaluated to determine the impact of the changed parameter on the SDL.

3.3.1 Selection of Key Parameters

The key parameters (including reactor power, core flowrate, core exit pressure, core inlet temperature, radial power distribution and axial peak magnitude and location) which significantly impact the calculation of DNBR used in the revised methodology are the same as those used in the previous SCD methodology.

As DPC stated, these key parameters associated with DNBR are generic to US PWRs and are independent of reactor design. Plant specific information determines the uncertainties associated with each parameter.

3.3.2 Selection of Uncertainties

As in the previous SCD formulation, in order to statistically combine the effect of the uncertainties of the parameters, DPC determined the uncertainties, uncertainty distributions and the uncertainty standard deviations. An uncertainty distribution is established for each of the seven variables with the nominal state conditions as the center. DPC's rational for assignment of uncertainty distribution was that a normal distribution was assumed when the uncertainty was due either to measurement uncertainty or a known statistical uncertainty distribution. Whenever such assumption could not be reasonably made, DPC chose the conservative approach of assuming a uniform distribution with estimated reasonable upper and lower bounds.

In addition to the seven variables related to the core and fuel conditions, two other variables related to the analysis method are assumed to impact computation of DNBR; code/model uncertainty and CHF correlation uncertainty. The uncertainties associated with the code/model allows for uncertainties due to the thermal hydraulic code and VIPRE core models.

The licensee stated in the topical report that the uncertainties and distributions will be justified on a plant-specific basis in the reload report for the first application of this methodology.

3.3.3 Propagation of Uncertainties

In order to combine the uncertainties to compute an overall DNBR uncertainty, a Monte Carlo method analysis is performed using the distribution of uncertainties defined with each variables. A Monte-Carlo computation is used to select sets of values at random (weighted by the distribution functions) about a statepoint of interest selected from a list of statepoints which form the basis for the statistical design limit.

3.3.4 Calculation of Statistical Design Limit (SDL) for DNBR

Using the Monte-Carlo generated input for the DNB computation, VIPRE-01 is run to calculate the DNBR for each case in a statepoint. The statistical DNB evaluations are performed at two levels. The first level of evaluation taking 500 propagated cases per statepoint is used to determine the DNB behavior over the entire analysis space. The second group of statepoints have 3000 cases each and contains a selected subset of the first group used to evaluate the statistical DNBR values and to improve the associated variance. Statistical analysis is then performed on the set of MDNBRs so generated to determine the statistical design limit (SDL) to replace the traditional DNB limit.

The statistical design limit is determined from the largest coefficient of variation based on the DNBRs computed by the Monte Carlo computations referred to above which avoid DNB at a 95% probability/95% confidence level.

3.3.5 Flexibility of the Methodology

DPC selected a few cases to demonstrate the flexibility of the methodology to changes in any of the key parameter uncertainty distribution, fuel designs or statepoint conditions.

The methodology is direct and general enough to be widely applicable to any fuel or reactor provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the CHF correlation and the uncertainties and associated distributions are reasonable.

3.3.6 Statistical DNB Behavior and Use of Two DNBR Limits

From the cases run with the above described method, DPC observed a dependence of statistically determined DNBRs on the axial location of the power peak, the magnitude of that peak and DNB location. When examining a series of calculations with the peak located in the lower 2/3 of the core and what DPC characterized as "flatter" power profiles, DPC observed a non-linear relationship between the DNBR responses and the axial power peak location and magnitude of the power peak. The study also indicated, for those cases, that the predicted limiting SDLs involved the DNBR occurring at the end of channel. DPC concluded that higher sensitivity of DNBR to certain key parameters was accompanied by higher SDL for the statepoints selected.

Observing the DNBR sensitivity and coupling that to the fact that Chapter-15 type analyses are all performed with power profiles which do not yield DNBRs near the limit, DPC proposed to divide the continuous DNBR space into two regions by the degree of predicted DNBR sensitivity to the axial power distribution: (1) One region contains the DNBRs predicted using the end-of-channel MDNBR limited axial power distributions (flatter and bottom-peaked) and (2) the other region contains all others. Correspondingly, DPC calculated separate statistical design limits for these regions. In the region associated with the flat and bottom-peaked power distributions, the predicted SDL was higher and a lower SDL value was computed in the other region. The higher limit is the one which would be used if the traditional one-limit methodology is to be approved.

The net result of DPC's proposed double limit would be that the lower limit would be used in all Chapter-15 type analyses. The method for determining the line separating the two areas has a fundamental impact since use of the lower limit results in a large gain in the margin.

However, the space to be divided is a continuous space, and there is a gradual transition from the region of higher DNBR limit to the region of lower DNB limits. DPC presented no definitive analytical method for dividing this space. Furthermore, any subdivision could result in reduction of the SDL from the current bounding DNBR limit and the result would be non-conservative.

4.0 CONCLUSION

The subject topical report together with DPC responses were reviewed and found to be acceptable with respect to documentation of the statistical core design methodology using the VIPRE computer code subject to the following limitations and restrictions:

1. The DPC developed statistical core design (SCD) methodology, as described in the submittal, is direct and general enough to be widely applicable to any PWR core, provided that the VIPRE-01 methodology is approved with the use of the core model and correlations including the CHF correlation subject to the VIPRE SER conditions. Furthermore, DPC must demonstrate that DPC's use of specific uncertainties and distributions based upon plant data and its selection of statepoints used for generating the statistical design limit are appropriate.

2. This methodology is approved only for use in DPC plants.

3. For the reasons set forth in Section 3.3.6, it is recommended that (i) of the two DNBR limits, only the use of the single most conservative DNBR limit be approved and (ii) the use of two SDLs not be approved at this time and that it be handled as a separate issue to be resolved in the future.

5.0 REFERENCES

1. "Duke Power Company Thermal-Hydraulic Statistical Core Design Methodology," DPC-NE-2005P, September 1992.

2. Letter from H.B. Tucker (DPC) to USNRC, Thermal/Hydraulic Statistical Core Design Methodology, DPC-NE-2005," September 29, 1993.

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4. "Acceptance for Referencing of Licensing Topical Report VIPRE-01: A Thermal-Hydraulic Code for Reactor Cores, EPRI NP-2511-CCM Vols. 1-4," May 1, 1986.

5. "McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01," DPC-NE-2004P-A, December 1991.

6. Letter from H.B. Tucker (DPC) to USNRC, submitting "Handouts Presented in the October 7 & 8, 1991 Meeting with NRC Staff and Contract Reviewers," October 16, 1991.

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Topical Report Main Body

ABSTRACT

This report presents Duke Power Company's methodology for performing statistical core thermal-hydraulic analyses. This method uses the models and thermal-hydraulic code currently approved for the Oconee and the McGuire/Catawba Nuclear Stations. The analyses method is based on DNBR limits that statistically account for the effects on DNB of key parameters such as reactor power, temperature, flow, and core power distribution. This report details the methodology development, the application to Duke plants, and the process for future technical enhancements and application to non-Duke reactors.

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Definitions

Case - A unique set of conditions analyzed by the thermal-hydraulic computer code. These conditions are based on a statepoint and include individual statistical variations of each key parameter.

Design DNBR Limit (DDL) - A numerical DNBR value that includes margin above the statistical design limit and is used for DNBR analyses. The DDL is calculated by multiplying the SDL by a fixed factor such as 1.10.

Key Parameter - A physical parameter that is important to the calculation of DNBR.

Statepoint - A unique set of fluid and reactor conditions evaluated for DNBR performance. These conditions include reactor power, pressure, temperature, coolant flow rate, and a three dimensional nuclear power distribution.

Statistical Core Design (SCD) - An analysis method that statistically combines the effects of all key parameter uncertainties associated with DNB predictions.

Statistical Design Limit (SDL) - A numerical DNBR value resulting from a SCD analysis that ensures, with a 95% probability at a 95% confidence level, DNB will not occur.

Statistical DNBR - The numerical value calculated by the SDL equation for a specific statepoint.

1.0 INTRODUCTION

The thermal-hydraulic design methodology accounts for the effects on DNB of the uncertainties of key parameters such as power, pressure, temperature and flow. Statistically combining these effects yields a better quantification of the DNB margin which, in turn, enhances core reload design flexibility. This report details the thermal-hydraulic statistical core design methodology developed by Duke Power Company for application to pressurized water reactors.

Several different statistical DNB analysis methods have been approved and are currently in use by various vendors and utilities. All the methods have slight differences but the major similarity is the basic concept that statistical behavior is defined by the sensitivity of DNB to key parameters and their associated uncertainties. When this relationship is well defined, a high degree of confidence in the applicability of the statistical DNB limit is assured.

1.1 CURRENT METHODOLOGY

The Thermal-Hydraulic Statistical Core Design (SCD) analysis method currently licensed for use by Duke Power Company is based on a Response Surface Model (RSM) prediction of DNBR behavior over a range of key parameters (Reference 3). The RSM is used to evaluate the impact of uncertainties on each parameter about a statepoint for a large number of cases. Figure 1 shows an overall process flowchart for the RSM based SCD analysis. This method has been approved by the NRC for use on the McGuire and Catawba Nuclear Stations.

1.2 REVISED METHODOLOGY

Duke Power Company has developed an alternative method to evaluate the statistical behavior of DNBR that both simplifies and enhances the accuracy of the original process. The simplified method uses the VIPRE-01 thermal-hydraulic computer code (Reference 1) to calculate the DNBR values for each set of reactor conditions. With this method, the intermediate step of developing and analyzing DNB response with the RSM is eliminated. Besides this enhancement, the overall process is identical to the currently approved methodology. Figure 2 shows the flowchart for the revised approach. Note that the major difference is the elimination of the first three steps shown in Figure 1. The revised methodology was used to determine the statistical design limit for three transient statepoints in Reference 3. Limited application of this methodology was reviewed and approved by the NRC for McGuire/Catawba thermal-hydraulic analyses as part of the review of Reference 3.

The revised SCD methodology is identical in most respects to other statistical thermal-hydraulic analysis methodologies. Key DNBR parameters are selected, their associated uncertainties are propagated about a statepoint, and a large number of DNBR's are calculated . The statistical behavior at that statepoint is evaluated by observing the distribution of the DNBR values and the mean and standard deviation of DNB for the given conditions. This same approach is repeated over a range of statepoints. The Statistical Design Limit (SDL) is based on the largest coefficient of variation and therefore the largest statistical DNBR value for the statepoints considered.

The statistical analysis method described in this report is applied to both the Oconee (Babcock and Wilcox) and McGuire/Catawba (Westinghouse) plant designs. The main body of this report details the specifics of the method and gives typical results. Two Appendices are included that contain plant specific information and results. This is necessary due to the differences in CHF correlations, fuel design, and specific uncertainties for each plant design. Appendix A contains the specific information for Oconee and Appendix B contains the same information for McGuire/Catawba. The plant specific thermal-hydraulic models and computer code configurations described in Reference 2 (DPC-NE-2003P-A) and Reference 3 (DPC-NE-2004P-A) are used in this analysis without modification.

This method of developing an SCD limit provides a more accurate representation of statistical DNB behavior because the thermal-hydraulic code is used directly to perform all DNBR calculations. Rather than relying on an algorithm such as the RSM, this methodology consists of over 151,000 individual VIPRE-01 cases at various statepoints. Because of the mechanistic approach used by this analysis, [

]

1.3 FUTURE USES

One benefit of the revised thermal-hydraulic analysis method is the ability to analyze factors outside of the original scope of analysis for a particular plant. This is due to the fact that the thermal-hydraulic code is used directly to determine statistical behavior. For example, if an assumed uncertainty should become non-bounding, the limiting statepoint can be re-evaluated to determine the impact of the changed parameter on the SDL. This method can also be used to evaluate a statepoint outside the range of the original key parameters assumed. If the statepoint statistical DNBR does not exceed the SDL, the statepoint can apply the licensed limit.

If the statepoint statistical DNBR does exceed the limit, appropriate measures, such as increasing the design DNBR limit (DDL) for that statepoint's analyses, can be used to ensure conservative DNBR limits are used. (The design DNBR limit approach is discussed in Section 2.5 of this report and Section 6.5 of Reference 3). This higher design limit will mean lower allowable radial power distributions for the affected statepoint. The higher limit would apply to all the subsequent analyses performed on that set of conditions. Another alternative to increasing the design DNBR limit is to use the available margin between the existing SDL and design DNBR limits to account for the change.

Secondly, this statistical analysis method shows generic DNB behavior that extends across fuel designs and plant types. The limiting SDL value is primarily affected by the particular Critical Heat Flux (CHF) Correlation used, the fuel assembly design, and the key parameter uncertainties. This allows the methodology to be applied to new or revised CHF correlations, new fuel assembly designs, or non-Duke plants, requiring only the submittal of an additional Appendix that provides the same information as included in the two attached.

2.0 STATISTICAL CORE DESIGN METHODOLOGY

The procedure for determining the statistical DNBR limit (SDL) contains four steps:

1. Selection of key parameters
2. Selection of uncertainties
3. Propagation of uncertainties
4. Calculation of the statistical DNBR limit (SDL).

The key parameters associated with DNBR are generic to pressurized water reactors and are independent of reactor design. The important plant specific information is the uncertainties associated with each parameter.

2.1 SELECTION OF KEY PARAMETERS

The key parameters used in this analysis are the same as those used in Reference 3 for SCD calculations. These are the parameters which significantly impact the calculation of DNBR and include:

- Reactor Power
- Core Flow Rate (including effects of core bypass flow)
- Core Exit Pressure
- Core Inlet Temperature
- Radial Power Distribution (including Hot Channel Factors)
- Axial Peak Magnitude
- Axial Peak Location

These seven parameters are used to set limits when performing reload thermal-hydraulic analyses. A statepoint in this analysis is a defined by a combination of all seven of these parameters.

The range of individual key parameter values in this analysis are based on statepoints that are using or will use the SCD DNB methodology. A majority of the statepoints analyzed have mean Minimum DNBR (MDNBR) values close to the statistical design limit itself. Table 1 shows typical statepoints that form the basis for the statistical design limit (Table 1 in the Appendices shows the statepoints analyzed for each plant). Table 4 in the Appendices contains the range of values for each key parameter represented by the analyzed statepoints.

Since this method mechanistically evaluates each statepoint, new or revised statepoints can be easily evaluated in the same manner. If, for example, the plant is uprated to a higher licensed power level or the pressure/temperature points change or a new transient statepoint is calculated, a propagation of the revised conditions about the limiting point would be performed. If the licensed SDL is conservative, no further action would be required. If the statistical DNBR value is higher, appropriate compensatory measures will be applied to ensure the allowable DNB behavior for the statepoint is conservatively bounded.

Duke Power's reload methodology, described in References 4 and 5, gives special attention to the axial power distribution (axial peak location and magnitude) in determining acceptable DNB performance. The axial peak location and magnitudes evaluated in this analysis are concentrated about a selected region. The axial power distribution area of interest is based on the peak magnitudes and locations that are typically predicted during the standard cycle design process. Figures 3A and 3B show a graphic representation of typical axial peak values (Fz) and locations (Z) calculated by the physics codes. Figure 3A is for Oconee and Figure 3B shows the same data for McGuire and Catawba.

2.2 SELECTION OF UNCERTAINTIES

A statistical core design analysis combines the effects of individual key parameter uncertainties that significantly affect DNB. Typical uncertainties for a reactor design are shown in Table 2 (Table 2 in each of the Appendices shows the plant specific values).

Distributions for the uncertainties are assumed to be either normal or uniform. The basis for the type of distribution assumed for each key parameter is included in the Appendices. Two additional uncertainties are included, one for the CHF correlation and one for code/model conservatism. The CHF correlation uncertainty is based on the standard deviation of the correlation data base and accounts for the correlation's uncertainty in DNB predictions. The code/model uncertainty allows for thermal-hydraulic code uncertainties and simplified versus detailed core model differences.

2.3 PROPAGATION OF UNCERTAINTIES

Multiple random cases are generated for each statepoint by independently varying all key parameters according to their associated uncertainty value and distribution. The SAS (Reference 6) statistical computer package random number function generators are used to create the necessary distributions. The key parameter distributions are calculated individually based on the type of uncertainty distribution and uncertainty magnitude.

There are two different types of uncertainties analyzed. The first type, denoted additive, is an uncertainty that has a fixed value. An example of this is the RCS temperature uncertainty of +/- 4 degrees F (see Table 2). The value is the same number of degrees F everywhere it is applied. The second type of uncertainty is called multiplicative and is based on a percentage of the parameter. An example of this is the radial power distribution uncertainty (3.25% in Table 2). Here, the radial peak used in each statepoint has an impact on the magnitude of the uncertainty. This statistical method of application accounts for both the uncertainty magnitude and distribution type (normal or uniform).

A total of either 500 or 3000 propagated cases (one case being a set of the seven key parameters) are generated for each statepoint. The different propagation sizes are compared to verify that the statistical behavior is consistent between the two levels of analysis and to be confident that the most limiting SDL is determined. Table 3 contains an example of key parameter propagations that together make up ten DNB cases for a given statepoint. The values were extracted from a typical 500 case propagation.

As stated previously, this analysis method allows for direct evaluation of the impact of increased uncertainties. If an uncertainty value assumed in the original analysis is exceeded in the future, the limiting statepoint can be re-analyzed with the changed value. If the statepoint statistical DNBR does not increase above the licensed limit, no further action is required. If it does, proper compensatory measures can be applied.

2.4 CALCULATION OF THE STATISTICAL DNBR LIMIT

After the VIPRE-01 code is used to calculate the MDNBR's for each case in a statepoint, the code/model and CHF correlation uncertainties are applied and the coefficient of variation (CV) is calculated as described in Reference 3. Cases that yield either a MDNBR value of less than 1.0 or that exceed the quality limit of the CHF correlation used are excluded from the data base prior to calculating the coefficient of variation. The distribution of MDNBR's is checked for normality by performing the D'Agostino (or D Prime) test on the final set of MDNBR values for each statepoint.

The appropriate Chi Square (Chi^2) and K factor (K) multipliers are determined based on the final number of MDNBR's for each statepoint. The statistical DNBR value for each statepoint is then calculated by the same equation as used in Reference 3,

$$SDL = 1.0 / \{1.0 - (K * Chi^2 * CV)\} \quad .$$

Table 4 contains example results of the mean, standard deviation, coefficient of variation, and the statistical DNBR values calculated for the Table 1 statepoints. {Table 3 in the Appendices contains the plant specific data.}

Table 4 contains two groups of statepoints in separate sections. This is because the statistical DNB evaluations in this analysis were completed at two levels. The first level of evaluation (500 propagated cases/statepoint) is used to determine the DNB behavior over the entire analysis space. The intent of the 500 case runs is to determine DNB behavior with respect to axial and radial peaking conditions, core power level, and changes in fluid conditions.

The second group of statepoints have 3000 cases each and are a selected subset of the first group (denoted by -T after the statepoint number). This group is used to determine the SDL of DNB analyses for each reactor type. Figures 4A (500 cases) and 5A (3000 cases) graphically show the results for Oconee at a selected set of fluid conditions. Figure 6A shows the comparisons of the same axial peak locations and magnitudes for different fluid conditions. Figures 4B, 5B, and 6B show the corresponding graphs for the McGuire/Catawba statepoints.

2.4.1 VARIANCE OF STATISTICAL DNB BEHAVIOR

Comparing all these Figures showing the statistical DNBR for [ ] across a range of fluid conditions and for different fuel/reactor types, a significant dependency [ ] is observed. [ ] show a more limiting statistical DNBR behavior than the remaining points. To evaluate this, the sensitivity of DNBR [ ] was evaluated in two manners.

First, the sensitivity of DNB [ ] was determined. This was done by [ ] constant and analyzing [ ] Figure 7A shows the sensitivity of DNB [ ] for the BWC correlation (Oconee). Figure 7B shows the sensitivity for the BWCMV correlation (McGuire/ Catawba).

Two items of interest are displayed in this representation. The first fact is that the slope [ ] Secondly, the slope in this area [ ] on the remainder of the graph. The absolute value of the slope is the important factor in determining the statistical response of a key parameter (slope is the sensitivity of DNBR [ ] This indicates that [ ] will have a different statistical behavior than the area where the slope is less steep. Note the agreement between Figures 7A and 7B (different fuel assembly designs and CHF correlations). This consistency continues to affirm that this observation is a mechanistic DNB behavior.

The second sensitivity evaluation varied all key DNB parameters of a statepoint by their uncertainty magnitude and calculated the slope for each (Δ DNBR / Δ parameter). These results are shown in Table 5. This type of analysis shows [ ]

Additionally, there is another phenomenon that is also present with [

]

This more limiting statistical behavior has been evaluated for generic applicability and was found to occur for each reactor type and CHF correlation as shown by Figures 4A, 4B, and 4C. Figure 4C is the same core geometry and statepoints as 4B but with the DCHF-1 CHF correlation (Reference 7). The statistical behavior [

] All these factors point to the conclusion that this more limiting statistical variance [

] is a generic, mechanistic DNB behavior and as such is applicable to any CHF correlation and core model (Oconee, McGuire, Catawba, etc).

2.4.2 FLEXIBILITY OF THE ANALYSIS METHOD FOR MODIFIED PARAMETER EVALUATIONS

Several different comparisons are included to demonstrate the ability of this method to address changes in core models or uncertainty distributions. Table 6 shows the results of three different evaluations. The first section includes two points that show the results of changing a single key parameter's uncertainty distribution from normal to uniform. Statepoints 33 and 34 from the McGuire/Catawba evaluation were identical in all respects except for the RCS flow distribution. In Statepoint 33, the distribution was normal (same for all other statepoints) and in Statepoint 34 the distribution was changed to uniform. The affects of this single parameter distribution change is readily calculated and shown to be negliable.

The section has two points that show the impact of a VIPRE-01 model change. Statepoints 37 and 38 both have identical conditions and uncertainties. Statepoint 37 used the eight channel McGuire/Catawba model from Reference 3 while Statepoint 38 used the fourteen channel model from Reference 8. Again, the comparison is easily accomplished and Table 6 shows the difference in the statistical DNBR values.

The third section contains a group of points that shows the comparison between Westinghouse OFA and Babcock Wilcox Mark-BW 17x17 mixing vane fuel. Four statepoints were run with both fuel types at the same fluid and power distribution conditions. The difference between the models is the changed subchannel flow areas, wetted and heated perimeters, gap connections, and grid form loss coefficients to correctly reflect each fuel type. The comparison shows that the OFA fuel model's behavior is the same as the Mark-BW model and the Mark-BW SDL conservatively bounds OFA fuel for McGuire and Catawba analyses.

2.4.3 FUTURE APPLICATIONS OF SCD METHODOLOGY

The fact that this analysis method is direct allows this statistical approach to be applied to any fuel type or reactor using an NRC approved thermal-hydraulic model and CHF correlation. Even if DNB behavior showed a stronger or weaker functionality for a different core design or CHF correlation, this method would correctly reflect this behavior in the statistical design limit or limits determined. If a new CHF correlation is used by Duke or if a different plant is analyzed, an additional Appendix will be submitted to the NRC detailing the model, CHF correlation, uncertainties, and statepoints used to determine the SDL for the plant specified.

2.5 APPLICATION OF THE SCD LIMIT

Since the statistical DNBR behavior demonstrated in this analysis shows [

]

The method for applying [

] Additionally, DNB analyses may be performed using a design DNBR limit (DDL) which includes margin above the statistical design limit [ ]

Should an analysis be performed that uses a new CHF correlation, for a non-Duke reactor, or for a new fuel design, statepoints [ ] will be analyzed to confirm the generic DNB behavior assumption and to determine the SDL [ ] This information will be reported to the NRC by submitting a new Appendix similar to Appendix A and B.

3.0 CONCLUSIONS

The methodology described in this report shows the major factors affecting statistical DNB behavior are [ ]

Since the statistical DNB behavior is controlled by these global parameters, [

]

This analysis method can be used to evaluate new fluid statepoints or revised uncertainties directly to determine the statistical limit. As long as the SDL is not exceeded, the established limits can be applied unmodified. If the statistical DNBR value for the new conditions is higher than the current limit, appropriate compensation measures such as increasing the design DNBR limit for the statepoint or using available margin between the design and statistical limits can be used. These actions penalize the statepoint by reducing the allowable radial peaking to ensure acceptable DNB behavior.

Since Duke's statistical thermal-hydraulic design methodology relies solely on DNB behavior, any PWR facility can be analyzed using this approach with an appropriate core model and bounding uncertainties. Also, new fuel designs or critical heat flux correlations can be evaluated to determine the appropriate SDL. The results of such an analysis would be submitted to the NRC for approval in the form of an additional Appendix that would contain the following:

1) Identification of the plant, fuel type, and CHF correlation with appropriate references to the approved fuel design and CHF correlation topicals.

2) Statement of the thermal-hydraulic code and model used with appropriate references to the approved code topical report.

3) A list of the key parameters, their uncertainty values, and distributions.

4) A list of the statepoints analyzed.

5) The Statistical Design Limits and how they are applied.

Table 7 contains a listing of some anticipated conditions and the corresponding actions.

4.0 SUMMARY

This report describes the analysis method used to determine the statistical core design DNB limit for reactor core thermal-hydraulic analyses. This methodology is used to account for the impact on DNB of the uncertainties of key parameters such as power, pressure, temperature, and core peaking. The methodology determines the statistical behavior of DNBR with respect to all these key parameters for many different statepoints and provides a method of applying the SCD DNB limits derived.

Duke has observed a significant statistical DNB behavior dependency [

] The

specific SCD DNB limits for the Oconee and McGuire/Catawba units are stated in the Conclusions section of the attached Appendices.

5.0 REFERENCES

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9. BWC Correlation Of Critical Heat Flux, BAW-10143P-A, Babcock and Wilcox, Lynchburg, Virginia, April 1985.

10. BWCMV Correlation Of Critical Heat Flux In Mixing Vane Grid Fuel Assemblies, BAW-10159P-A, Babcock and Wilcox, Lynchburg, Virginia, May 1986.

TABLE 1. Typical Reactor SCD Statepoints

Stpt #	Power	Pressure	Temperature	Flow	Axial Peak	FΔh
DNB Limit Line Statepoints						
1						
3						
4						
12						
14						
17						
26						
Loss Of RCS Flow Transient Statepoints						
21						
24						
29						
Uncontrolled Bank Withdrawal Transient Statepoint						
33						
Nominal Operating Statepoints						
16						
27						

TABLE 2. Typical Statistically Treated Uncertainties

Parameter	Uncertainty / Standard Deviation	Type of Distribution
Reactor Power	+/- 2% / +/- 1.22%	Normal
Core Flow		
Measurement	+/- 2.2% / +/- 1.34%	Normal
Bypass Flow	+/- 1.5%	Uniform
Pressure	+/- 30 psi	Uniform
Temperature	+/- 4 deg F	Uniform
$F^N_{\Delta H}$		
Measurement	+/- 3.25% / 1.98%	Normal
$F^E_{\Delta H}$	+/- 3.0% / 1.82%	Normal
Spacing	+/- 2.0% / 1.22%	Normal
F_Z	+/- 4.41% / 2.68%	Normal
Z	+/- 6 inches	Uniform
DNBR		
Correlation	+/- 16.78% / 10.2%	Normal
Code/Model	[ ]	Normal

TABLE 3. Typical Monte Carlo Propagation Statepoint Values

(Values After Uncertainty Propagation of Stpt. # 1 from TABLE 1)

Base Statepoint

Case#	Power	Press	Temp	Flow	Fz	Z	FΔh
0							

Propagation

Case#	Power	Press	Temp	Flow	Fz	Z	FΔh
1							
50							
100							
150							
200							
250							
300							
350							
400							
450							
500							

TABLE 4. Example of Typical Statepoint Statistical Results

Section 1 - 500 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
DNB Limit Line Statepoints				
1	[			
3				
4				
12				
14				
17				
26				]
Loss Of RCS Flow Transient Statepoints				
21	[			
24				
29				]
Uncontrolled Bank Withdrawal Transient Statepoint				
33	[			]
Nominal Operating Statepoints				
16	[			
27				]

TABLE 4 - continued Example of Typical Statepoint Statistical Results

Section 2 - 3000 Case Runs

Statepoint #	Mean	σ	Coefficient of Variation	Statistical DNBR
DNB Limit Line Statepoints				
3-T	[			
4-T				
12-T				
14-T				]
Nominal Operating Statepoint				
16-T	[			]

TABLE 5. Individual Key Parameter Slopes At Statepoint Conditions

Key Parameter*	Stpt 6	Stpt 25	Stpt 9	Stpt 21

[ ]

The statepoints listed above are from the McGuire/Catawba 500 case runs. [ ] Statepoints 6 and 25 [ ] Statepoints 9 and 21 [ ]

* All values shown are in %DNBR per unit of parameter (Δ DNBR / Δ parameter). For example, the first entry in the table of [ ] means a [ ] DNBR change for every 1% power change.)

Table 6. Uncertainty and Model Changes - Impact On Statistical DNBR Behavior

<u>Uncertainty Distribution Change</u>

The following two statepoints show the change in the statistical behavior for a fixed set of conditions if the RCS flow uncertainty distribution is changed from normal to uniform.

Statepoint #	RCS Flow Uncertainty Dist.	Coefficient Of Variation	Statistical DNBR
33	Normal		
34	Uniform		

<u>Thermal-Hydraulic Model Detail Change</u>

The following two statepoints show the change in the statistical behavior for a fixed set of conditions using two different VIPRE-01 models.

Statepoint #	McGuire/Catawba VIPRE-01 Model	Coefficient Of Variation	Statistical DNBR
37	8 Channel		
38	14 Channel		

<u>Minor Fuel Geometry and Design Changes</u>

The following eight statepoints show the change in the statistical behavior for the geometry and form loss coefficient changes between Mark-BW and OFA fuel assemblies for the same fluid and peaking conditions.

MARK-BW			OFA		
Statepoint #	Coefficient Of Variation	Stat. DNBR	Statepoint #	Coefficient Of Variation	Stat. DNBR
6			40		
12			41		
14			42		
16			43		

TABLE 7. SDL Evaluation And Re-Submittal Criteria

The following table lists different events or conditions that would require an evaluation of the applicability of an approved SDL and the subsequent actions based on the results of the analysis.

CONDITION	ACTION
Revised uncertainty larger than the limiting value used in the original analysis.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
Revised uncertainty distribution.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
New statepoint.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
Minor modifications to the current fuel design.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
A modified CHF correlation.	SDL ≤ Limit, No Action. SDL > Limit, Specifically compensate.
Change to a new fuel design/fuel type.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
A new CHF correlation.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
Duke analysis of a non-Duke reactor.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
New Thermal-Hydraulic Code.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.

FIGURE 1

RSM BASED SCD FLOWCHART

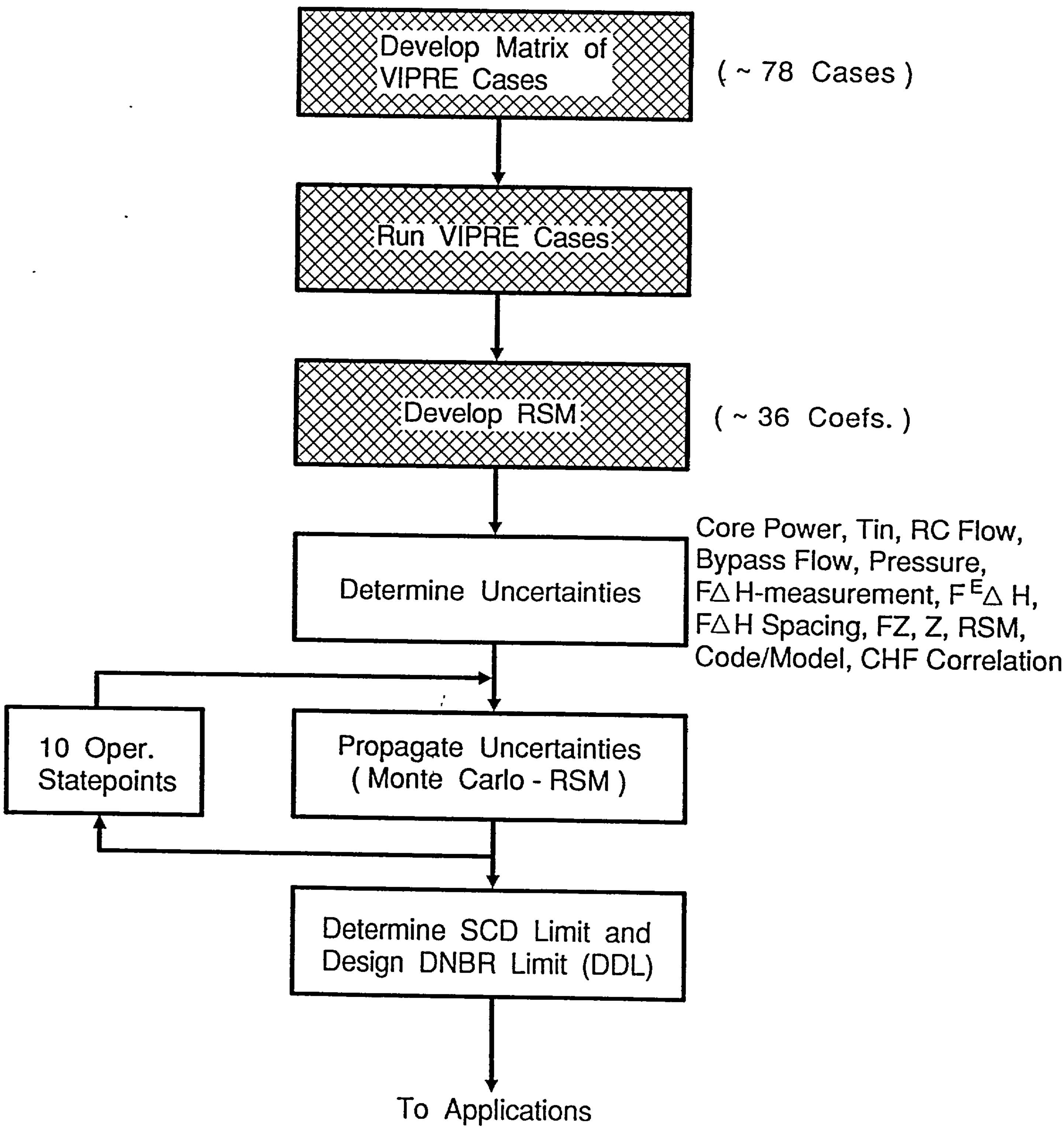

FIGURE 2

REVISED SCD FLOWCHART

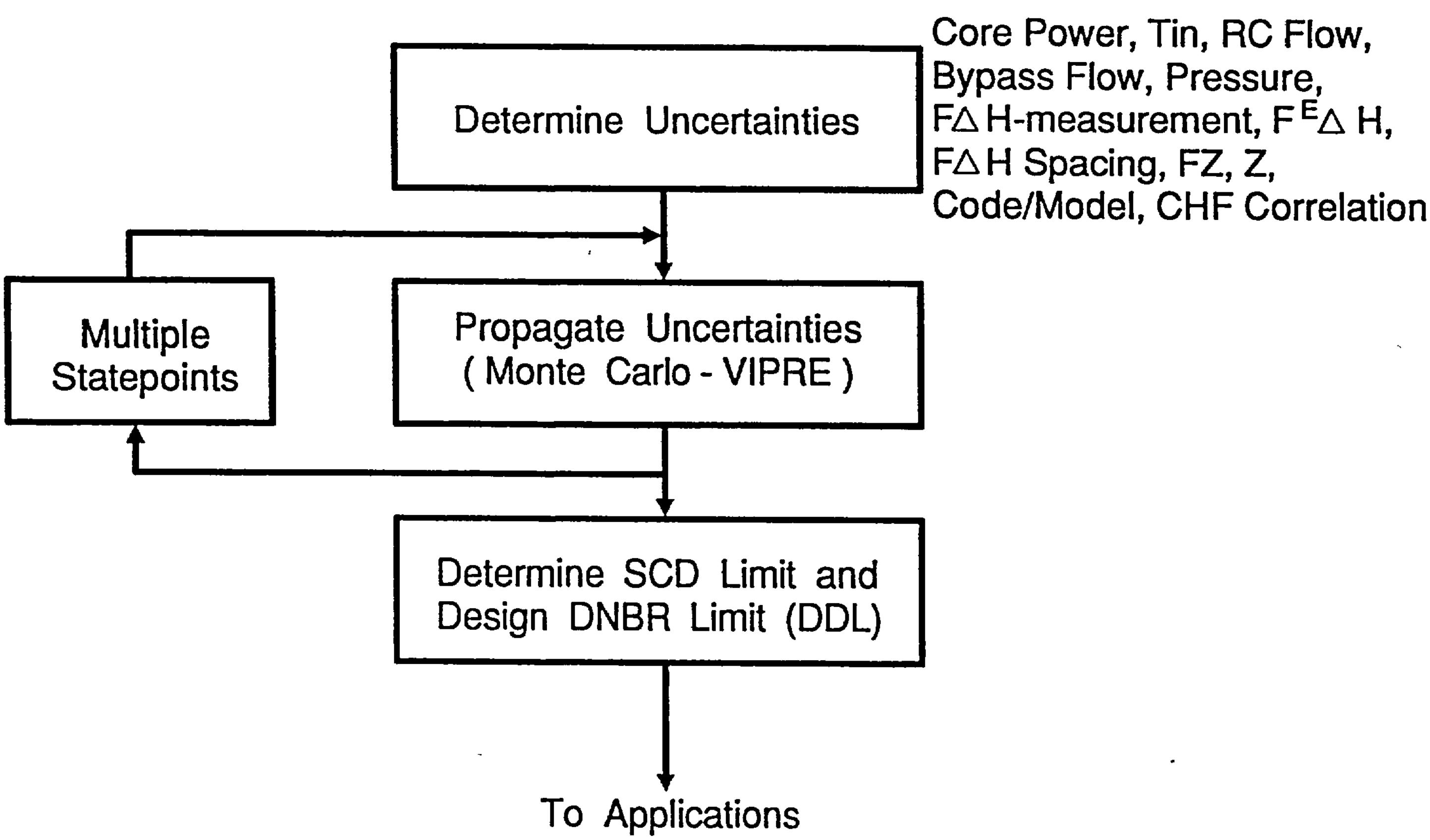

FIGURE 3A Oconee Physics Code Axial Power Distributions (Peak Magnitude and Locations)

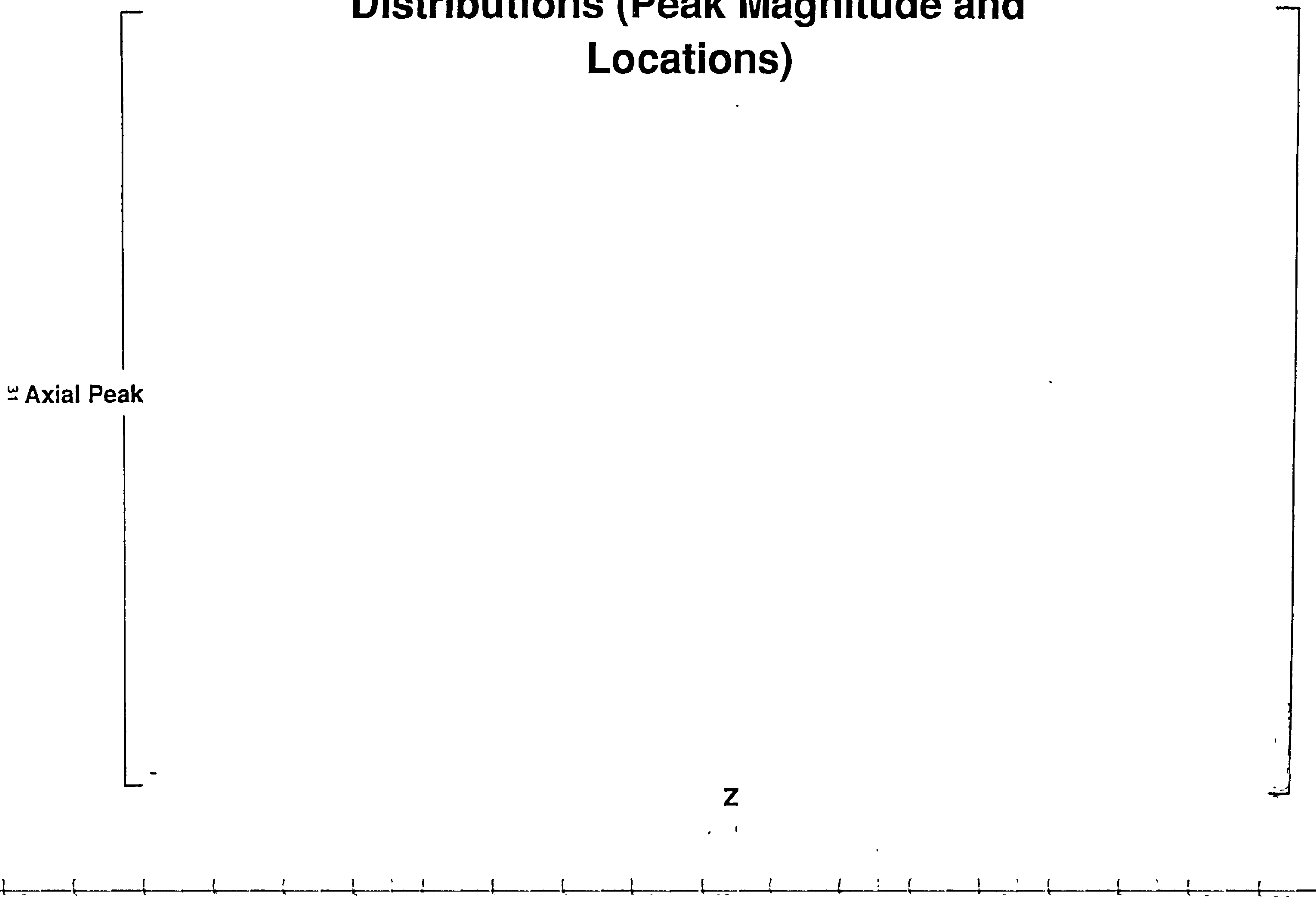

FIGURE 3B M/C Physics Code Axial Power Distributions (Peak Magnitude and Location)

Axial Peak

Z

FIGURE 4A
Oconee SDL Distribution At Constant Conditions, BWC
500 Case Propagations

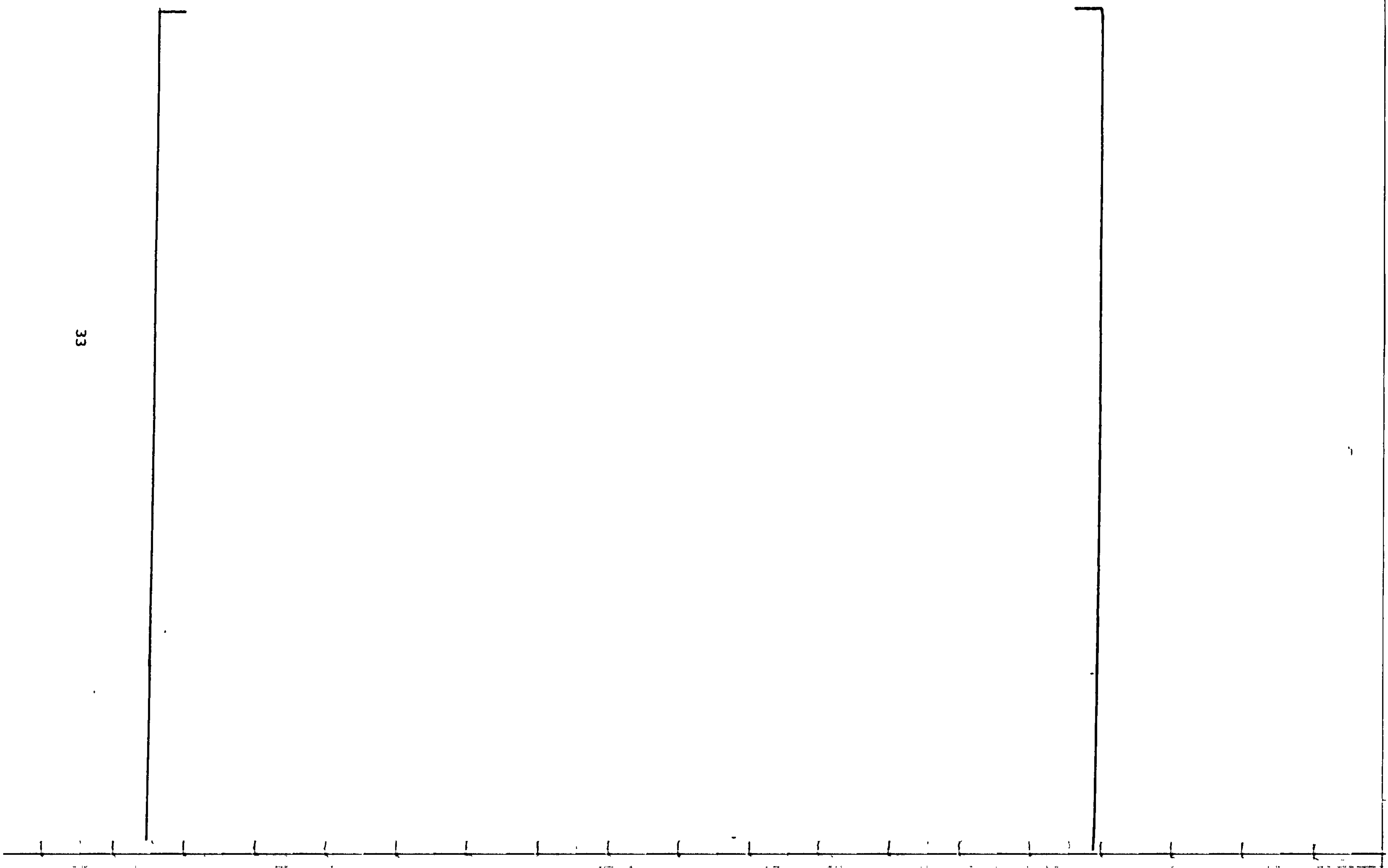

FIGURE 4B

M/C SDL Distribution At Constant Conditions, BWCMV

500 Case Propagations

FIGURE 4C

M/C SDL Distribution At Constant Conditions, DCHF-1

500 Case Propagations

FIGURE 5A
Oconee SDL's For 3000 Case Statepoints, BWC

FIGURE 5B

M/C SDL's For 3000 Case Statepoints, BWCMV

FIGURE 6A
Oconee SDL's For Various Conditions

FIGURE 6B
M/C SDL's For Various Conditions

FIGURE 7A Sensitivity of DNBR
BWC

FIGURE 7B Sensitivity of DNBR[ ] BWCMV

FIGURE 8A

BWC

FIGURE 8B

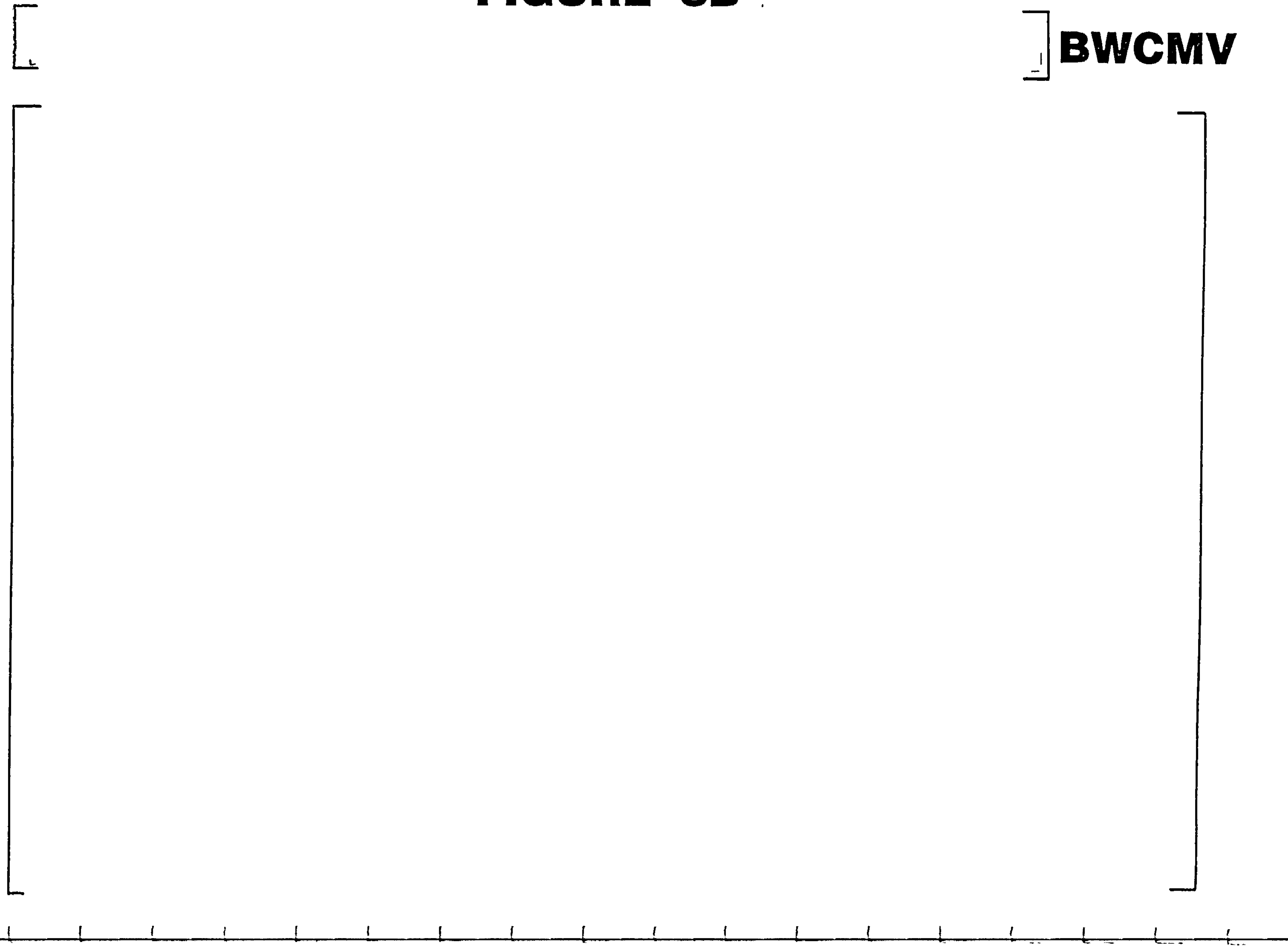

FIGURE 9

Example

Application

Attachment 1
Response To Request for Additional Information For Main Body, Appendix A, and Appendix B

Revision 0

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DUKE POWER

September 29, 1993

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D. C. 20555

Subject: Oconee Nuclear Station
Docket Nos. 50-269, -270, -287
McGuire Nuclear Station
Docket Nos. 50-369, -370
Catawba Nuclear Station
Docket Nos. 50-413, -414
Thermal/Hydraulic Statistical Core Design Methodology,
DPC-NE-2005

By letter dated September 28, 1992, Duke Power Company submitted Topical Report DPC-NE-2005, "Thermal/Hydraulic Statistical Core Design Methodology." The NRC staff issued a request for additional information (RAI) dated July 27, 1993. Attached are the responses to the questions contained in the RAI.

In accordance with 10 CFR 2.790, Duke Power Company requests that the attached information relating to DPC-NE-2005 be considered proprietary. Information supporting this request is included in the affidavit which appears as Attachment I.

If we can be of assistance in your review please call Scott Gewehr at (704) 382-7581.

Very truly yours,

M.S. Tuckman

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File: GS-801.01

ATTACHMENT I

AFFIDAVIT OF M. S. TUCKMAN

1. I am Senior Vice President, Nuclear Generation Department, Duke Power Company ("Duke"), and as such have the responsibility of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear plant licensing, and am authorized to apply for its withholding on behalf of Duke.

2. I am making this affidavit in conformance with the provisions of 10 CFR 2.790 of the regulations of the Nuclear Regulatory Commission ("NRC") and in conjunction with Duke's application for withholding which accompanies this affidavit.

3. I have knowledge of the criteria used by Duke in designating information as proprietary or confidential.

4. Pursuant to the provisions of paragraph (b)(4) of 10 CFR 2.790, the following is furnished for consideration by the NRC in determining whether the information sought to be withheld from public disclosure should be withheld.

   (i) The information sought to be withheld from public disclosure is owned by Duke and has been held in confidence by Duke and its consultants.

   (ii) The information is of a type that would customarily be held in confidence by Duke. The information consists of analysis methodology details, analysis results, supporting data, and aspects of development programs, relative to a method of analysis that provides a competitive advantage to Duke.

   (iii) The information was transmitted to the NRC in confidence and under the provisions of 10 CFR 2.790, it is to be received in confidence by the NRC.

   (iv) The information sought to be protected is not available in public to the best of our knowledge and belief.

   (v) The proprietary information sought to be withheld in this submittal is that which is marked in the proprietary version of the report DPC-NE-2005, "Thermal/ Hydraulic Statistical Core Design Methodology" and supporting documentation, and omitted from the non-proprietary versions.

M. S. Tuckman

M. S. Tuckman

(continued)

AFFIDAVIT OF M. S. TUCKMAN

This information enables Duke to:

(a) Respond to Generic Letter 83-11, "Licensee Qualification for Performing Safety Analyses in Support of Licensing Actions."

(b) Respond to NRC requests for information regarding the transient response of Babcock & Wilcox and Westinghouse reactors.

(c) Support license amendment and Technical Specification revision request for Babcock & Wilcox and Westinghouse reactors.

(vi) The proprietary information sought to be withheld from public disclosure has substantial commercial value to Duke.

(a) It allows Duke to reduce vendor and consultant expenses associated with supporting the operation and licensing of nuclear power plants.

(b) Duke intends to sell the information to nuclear utilities, vendors, and consultants for the purpose of supporting the operation and licensing of nuclear power plants.

(c) The subject information could only be duplicated by competitors at similar expense to that incurred by Duke.

5. Public disclosure of this information is likely to cause harm to Duke because it would allow competitors in the nuclear industry to benefit from the results of a significant development program without requiring a commensurate expense or allowing Duke to recoup a portion of its expenditures or benefit from the sale of the information.

M. S. Tuckman

M. S. Tuckman

(continued)

AFFIDAVIT OF M. S. TUCKMAN (Page 3)

M. S. Tuckman, being duly sworn, on his oath deposes and says that he is the person who subscribed his name to the foregoing statement, and that the matters and facts set forth in the statement are true.

M. S. Tuckman

M. S. Tuckman

Sworn to and subscribed before me this 29th day of September, 1993. Witness my hand and official seal.

Linda L. Kessler

Notary Public

My commission expires May 7, 1994.

LINDA L. KESSLER
NOTARY PUBLIC

Request for Additional Information and Responses To Topical Report DPC-NE-2005P

The questions are shown in italics and the responses immediately follow.

1. *Explain DPC's intent for this topical report. Does DPC seek its review with respect to its plants or generic PWR application? How does DPC plan to deal with the restrictions and requirements imposed by the VIPRE-01 code SER?*

The intent of this submittal is to outline a statistical Departure from Nucleate Boiling methodology. In DPC-NE-2005, DPC has outlined a statistical analysis method that is based on inherent behavior of the DNBR phenomena in pressurized water reactors. The numerical value of the Statistical Design Limit (SDL) will vary, depending on the CHF correlation used and parameter uncertainties assumed. However, direct use of the VIPRE-01 thermal hydraulic code (rather than the RSM) to calculate the phenomenological statistical variance of DNBR insures the direct applicability of this method to many varying fuel designs and parameter conditions.

DPC seeks the following approval from the NRC regarding this report:

1) Review and approval of the methodology and the stated statistical DNB limits for use at Oconee, McGuire, and Catawba based on the information in the body of the report and the site specific information in the Appendices.
2) Review and approval of the use of the methodology for future analyses of non-DPC reactors consistent with the commitments made in Section 1.3 and 2.5 of the report. This involves development or justification of the models and uncertainties used for any other site. If DPC were to extend this method to another PWR facility, a separate submittal will be made detailing the intent and justification for specific modeling assumptions, choice of flow models and correlations, and plant specific input data, as well as the resulting statistical DNB limits. The form of this submittal would be an additional Appendix to this report. This meets item (3) of Section 3 of the VIPRE-01 SER. The SDL would be calculated using the methodology outlined in the body of the report.

2. *DPC previously submitted two sets of DNB models for each type of plant. One was approved for use in steady-state type calculations and the other for use in transient type calculations. Since there are differences between these models on the basic level of model/input selection, discuss the impact of these differences on SDL determined. The SCD is developed based upon a series of steady state calculations. Explain how the SDL is used for transient analysis.*

Both models used by DPC were included in the statistical propagations detailed in the report. This is explained on page 14 of the report. Statepoints 37 and 38 in Appendix B are identical in fluid and peaking conditions. Statepoint 37 was propagated with the eight channel M/C model from Reference 5 and Statepoint 38 used the fourteen channel model from Reference 6. Table 6 of the report (page 27) shows the results of this comparison. The difference in Statistical DNBR's is negligible.

The determination of whether the SCD limit can be used for a transient is based on the fluid conditions at the point of minimum DNBR (MDNBR) during the transient. If the power, pressure, temperature, and flow rate of this statepoint fall within the parameter range listed in Table 4 of the appropriate Appendix, the SDL can be used. All the statepoint statistical propagations are made from a single set of fluid and peaking conditions.

3. Discuss how "appropriate compensatory measures" will be applied to ensure the allowable DNB behavior for the statepoint is conservatively bounded.

Please refer to the Definitions page (Page V) of the report for the following definitions:

Design DNBR Limit, DDL
Statistical Design Limit, SDL
Statistical DNBR

The term statistical DNBR applies to a specific statepoint, the SDL is the licensed limit, and the DDL is the MDNBR value used in steady-state and transient DNB analyses.

As stated in the report, new statepoints or revised uncertainties can be evaluated directly with this method. As long as the statistical DNBR value is less than the SDL, the statepoint is conservatively bounded by the SDL. If, however, the statistical DNBR value is larger than the SDL, actions can be taken to ensure that DNB predictions for the condition meet the required 95/95 acceptance criteria. These compensatory measures include either

1) Increasing the design DNBR limit for the statepoint.
2) Using available margin present between the statistical and design DNB limits (between the SDL and the DDL).

Increasing the design DNBR limit (DDL) will increase the minimum DNBR that is allowed in the analysis of the statepoint This requires another key transient analysis input (such as maximum allowable peaking) to be reduced. Penalizing a statepoint in this manner will ensure the required DNBR protection is maintained.

Another equally valid method is to apply any unused margin already available between the statistical (SDL) and design DNBR limit (DDL). This margin is inherently retained in all analyses by using the DDL in design calculations which includes margin above the SDL. A portion of this margin is currently used to account for such things as reactor vessel flow anomalies, instrumentation biases that cannot be statistically compensated for, and physical changes to the fuel assembly not accounted for in standard models (such as rod bow). The margin remaining after all of the DNBR penalties are accounted for can be used to compensate for the increase in SDL required for a particular statepoint. Either of these methods will conservatively adjust the MDNBR limit that must be met in the analysis to ensure adequate protection is maintained.

4. *Explain why RCS flow is varied only between 100 and 106.5% and not below 100% (see Table A-1 on p. A-3) even for low flow cases.*

The percent flow listed for the low flow cases in Table A-2 is in error. The flow rate used for all the statepoints identified as Low Flow in the Comments column was [ ] Additionally, the Minimum flow value listed on Table A-4 for percent design RCS flow should be [ ] The corrected pages are included with this response.
Additionally, the flow chart in Figure 2 also contains a typo. The Propagate Uncertainties box should have the words (Monte Carlo - VIPRE) underneath. The RSM is not used at all in the revised method described by the report. A corrected page 30 is also included.

5. *Explain thoroughly how ranges of uncertainties and their associated standard deviations are determined.*

The numerical range of each uncertainty is selected to bound the value calculated for the parameter. This ensures that conservative statistical behavior is calculated and allows for changes in the uncertainty value without requiring re-analysis of the SDL.

(a) *Explain how uncertainties in instrumentation are accounted for. What is meant by the term "random uncertainty" (see Table A-2)? Explain how it is related to instrument error uncertainty.*

The term "random uncertainty" used in Table A-2 of the report means the instrument uncertainties such as sensor calibration accuracy, rack drift, sensor drift, etc., that are combined by the SRSS method. The term was used because the biases which are constant in sign (either positive or negative) are not included in the propagation of an uncertainty and must be accounted for by another means, such as a DNB penalty.

(b) *Identify the sources of the quantitative ranges of uncertainties and their associated standard deviations (for both types of plants).*

The source of the quantitative ranges and the standard deviations are provided on Table 1 of this response for each plant. The statistical propagations for each normally distributed parameter are based on the standard deviation numerical values. Uniform uncertainty propagations are based on the uncertainty numerical magnitude.

6. Explain thoroughly the mechanistic DNB behavior observed in Figures 7A and B.

Figures 7A and 7B in the report show the sensitivity of DNBR to axial peak location and magnitude. This sensitivity was calculated by holding all other parameters (power, pressure, temperature, flow, and radial peaking) constant. Both the BWC (7A) and BWCMV (7B) CHF correlation results are shown. These graphs show that the response of DNBR varies with axial peak conditions.

(a) Discuss why the sensitivity to the axial peaks and locations is significantly stronger for Oconee than it is for M/C.

The evaluations contained in the report indicate that the numerical value of the SDL is dependent on the CHF correlation used in the analysis. Table 5 in the report contained individual parameter sensitivities to DNB for the BWCMV CHF correlation in both axial peak areas defined. Table 2 in this response contains an identical sensitivity evaluation for the BWC and DCHF-1 CHF correlations in both axial peak areas.

For the region of higher statistical behavior, comparison of the BWC and BWCMV sensitivities shows the sensitivity calculated for each key parameter with the BWC correlation has slightly higher sensitivity to DNBR. This results in a higher final calculated SDL. The sensitivity values are more consistent when the same evaluation is made in the lower SDL area and the corresponding statistical DNBR's for the two correlations are almost identical. Correspondingly, the DCHF-1 correlation has lower sensitivities in both areas and has the lowest statistical DNBR in both cases.

Again, Table 2 in this response as well as Table 5 in the report (page 26) shows that the behavior is remarkably consistent between Oconee and McGuire/Catawba and is linked to axial power distribution. There is a difference in the numerical value of the statistical DNBR, and the key to this is the CHF correlation being used. DPC's conclusion is that the general behavior is mechanistic and this is proven by the consistent behavior when the sensitivity is calculated for different fuel types (15x15 non-mixing vane and 17x17 mixing vane), different fuel vendors (Westinghouse and Babcock & Wilcox), and even different CHF correlations (BWC, BWCMV, and DCHF-1).

(b) DPC's conclusion based upon Figure 6A and B on p. 13 is not clear. Explain further.

The discussion on page 13 and Figures 4A, 4B, 4C, 5A, 5B, 5C, 6A and 6B of the report show how the statistical DNB behavior is much more dependent on axial peak location than on the fluid parameter values for a particular statepoint, the fuel type, or the CHF correlation. The Figure 4 and 5 series show how the statistical DNB behavior changes with shifts in the axial power distribution. The axial peak location has a large impact on the statistical DNB value. By contrast, Figures 6A and 6B show how little the statistical DNB behavior changes with large changes in the statepoint pressure, temperature, flow rate, and core power variables. This means that if the SDL is determined in either of the axial power distribution areas for one set of fluid conditions, this SDL value would be consistent even if the fluid conditions changed dramatically.

7. *Provide a table which identifies which DNB methodology is used for each transient and explain each such selection.*

The McGuire/Catawba DNB transients currently analyzed using the SCD methodology are listed in Table 3 of this response. No transients are currently analyzed for Oconee with the SCD methodology. All of the transients analyzed with the SCD methodology were selected based on the values of the individual parameters at the point of MDNBR during the transient as explained in the response to Question 2. If these values are within the range for each parameter defined on Table 4 of the appropriate Appendix, the SCD limit can be applied to the transient.

As discussed by the note below Table 4-A and 4-B, this parameter list is subject to change. One of the advantages of the explicit evaluation method describe in the report is the ability to specifically evaluate new conditions for SCD limit applicability. If a new statepoint has a parameter(s) outside the given range, it would be analyzed and if the current SCD limit is conservative, the table would be updated to show the expanded range. The transient that generated the statepoint would then be included on the internal DPC list (Table 3 of this response). This increased parameter range would not be reported directly to NRC.

8. *Explain the last two paragraphs of Section 2.4. Discuss the need to perform statistical DNB analysis in two levels and with two different sample sizes.*

The two different sample sizes were used to minimize the total number of cases propagated for each set of fluid conditions analyzed. The first level of 500 cases per statepoint is used to quickly evaluate the behavior of a statepoint with respect to the two axial peak areas. This shows the statistical DNB behavior and approximate numerical SDL value for the fluid conditions being evaluated.

The second group of 3000 case statepoints are selected to calculate the limiting SDL value for the reactor type being analyzed. The increase in number of cases to 3000 provides a more thorough evaluation of the statistical DNB response and improves statistically the Chi Square and K factor multipliers used to conservatively increase the coefficient of variation in the final SDL calculation. The licensed statistical design limit is greater than the largest value calculated in all the 3000 case propagations for each axial peak area.

DPC may increase the number of cases at a particular statepoint for future evaluations to take advantage of the improved effect on the statistical multipliers. This increase in the number of cases is consistent with the methodology as presented and does not in any way reduce the conservatism of the SDL limit calculated. Increasing the number of cases simply reduces the statistical uncertainty associated with calculation of the coefficient of variation.

9. Explain the rational for and appropriateness of selection of certain sets of statepoints to determine the impact of changes on statistical DNBR behavior (see Table 6).

The evaluations in Table 6 of the report show how little the statistical DNB behavior is affected by small modifications in the analysis. The first section shows the change for identical conditions and models with a change in one parameter uncertainty distribution (normal versus uniform). Section 2 shows the change if a different VIPRE-01 model is used with the same fluid conditions, peaking conditions, and uncertainty distributions. The last section shows the change with the same VIPRE-01 model, fluid conditions, and uncertainties but with a different fuel design. As discussed in the response to question 6b, Figures 6A and 6B demonstrate the there is very little change in statistical DNB behavior for large changes in the statepoint pressure, temperature, flow rate, or core power variables. Thus, the sensitivity of the SDL to other changes can be evaluated using a single statepoint.

All of these evaluations were included to further demonstrate that the statistical DNB behavior and SDL are more closely related to the CHF correlation and axial power distribution than to small perturbations in individual uncertainties, VIPRE-01 models, or fuel type. This evaluations also provide the basis for the criteria for re-submittal or in-house evaluation detailed on Table 7 (as explained in the response to Question 10).

10. Explain Table 7.

Table 7 in the report is intended as a guide for use by DPC in evaluating what action must be taken for anticipated changes (a revised uncertainty, new fuel type, etc.). In all cases, the evaluations will use the methodology detailed in the report. Basically, changes that are anticipated to have a negligible or very small impact on the SDL will require internal DPC evaluation. Only changes that have a significant impact on the calculated SDL number will be submitted to the NRC for approval.

An example of the kind of anticipated events is a change in an uncertainty magnitude. For this instance, limiting SCD statepoints in each axial power distribution area will be evaluated to determine the impact on the SCD limit. If the statistical DNBR value is the same or smaller than the SDL, no additional work is required. If the value is larger, appropriate compensation measures will be used to conservatively compensate for the change (as described in the answer to Question 3). This same approach will be used for different uncertainty distributions, new fluid or peaking condition statepoints, or minor modifications to the fuel assembly design.

For changes that will have a much bigger impact on the statistical DNB behavior, the impact of the change will be evaluated and a new Appendix to this report submitted for NRC approval. This additional Appendix will have the same format and content as the two already included in the report. Examples of when this approach would be used are a completely new fuel assembly design, a new thermal hydraulic code, a new CHF correlation, or DPC analysis of a third party's reactor.

A slight change to Table 7 is also included in the response to this question. The original table required that a modified CHF correlation would require submittal of a new Appendix. This has been changed to require an evaluation only. The term modified means the form of the CHF correlation is the same, just a single factor or multiplier has been changed or added. This change is because a modified correlation will not impact the statistical DNB behavior and will not significantly change the SDL compared to the original correlation. A modified correlation will still require a separate CHF correlation topical submittal to the NRC. Any other changes that affect the correlation form will be considered a new CHF correlation.

11. Provide the SDL if no distinctions are made of axial power distributions.

The results of the entire analysis completed in the report show how mechanistic the statistical DNB response is to axial power distribution. This mechanistic behavior was determined by direct use of the thermal hydraulic codes, models, and correlations used in DNB predictions. This behavior is consistent with different fluid conditions, fuel geometries, and CHF correlations. The one consistent fact is the larger statistical variation for a specific set of axial peaks. The use of two statistical DNB limits to address this behavior is a straight forward application. Use of a single limit would be unnecessarily conservative. However, if the appropriate distinctions are not made for the generic DNB behavior with axial power distributions, the SDL for all cases will be the largest value calculated for all the conditions evaluated. If this restriction were imposed, the SDL would be 1.43 for Oconee and 1.40 for McGuire/Catawba.

TABLE 1

Uncertainty Ranges And Standard Deviations

The following table shows the source of the quantitative range of each uncertainty and its associated standard deviation. Section 1 of the table contains the Oconee information and Section 2 the sources for the McGuire/Catawba values.

SECTION 1 - Oconee

Parameter	Source
Power	Standard deviation of 1.0% based on DPC calculations. Uncertainty value is a 2σ value (2%).
Pressure	Standard deviation of 15 psi based on DPC calculations. Uncertainty value is a 2σ value (30 psi).
Temperature	Standard deviation of 1.0 degrees Fahrenheit based on DPC calculations. Uncertainty value is a 2σ value (2 deg F).
Flow	Standard deviation of 1.0% design flow based on DPC calculations. Uncertainty value is listed as a 2σ value (2%).
$F\Delta H$	Standard deviation of 2.84% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for Oconee (Reference 1).
F_Z	Standard deviation of 2.91% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for Oconee (Reference 1).
Z	Uncertainty range of +/- 6 inches. Selected based on the noding size of nuclear codes. No standard deviation (uniform uncertainty).
Local Heat Flux HCF	Uncertainty range of [ ] Based on calculated values from the nuclear fuel vendor. Standard deviation is calculated from [ ] uncertainty value [ ]

Rod Power HCF	Uncertainty range of [ ] Based on calculated values from the nuclear fuel vendor. Standard deviation is calculated from [ ] uncertainty value [ ]
Hot Channel Flow Area	Uncertainty range of [ ]. Based on calculated values from the nuclear fuel vendor. No standard deviation (uniform uncertainty).
CHF Correlation	Standard deviation of 8.88% calculated from the BWC CHF test data base (Reference 2).
Thermal Hydraulic Code / Model	Uncertainty range of [ ] Value used in Reference 3. Standard deviation is calculated from the [ ] uncertainty value [ ].

SECTION 2 - McGuire/Catawba

Parameter	Source
Power	Uncertainty Range of 2%. Selected from Reference 5. Kept at 2% to bound specific uncertainties calculated for M/C. Standard deviation is calculated from 2% uncertainty value (2/1.64 = 1.22%).
Pressure	Uncertainty Range of 30 psi. Selected from Reference 5. Kept at 30 psi to bound specific uncertainties calculated for M/C. No standard deviation (uniform uncertainty).
Temperature	Uncertainty Range of 4 degrees Fahrenheit. Selected from Reference 5. Kept at 4 degrees to bound specific uncertainties calculated for M/C. No standard deviation (uniform uncertainty).
Flow	Uncertainty Range of 2.2%. Selected from Reference 5. Kept at 2.2% to bound specific uncertainties calculated for M/C. Standard deviation is calculated from 2.2% uncertainty value (2.2/1.64 = 1.34%).
$F\Delta H$ Measurement	Standard deviation of 1.98% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for M/C (Reference 1).

Engineering HCF	Uncertainty range of 3.0%. Selected based on the value in Technical Specifications. Standard deviation is calculated from the 3% uncertainty value (3/1.64 = 1.82%).
Spacing	Uncertainty range of 2.0%. Selected from Reference 5. Standard deviation is calculated from the 2% uncertainty value (2/1.64 = 1.22%).
F_Z	Standard deviation of 2.68% calculated based on the nuclear code packages used for core design and analysis. This standard deviation bounds the highest value from the code package combination used for M/C (Reference 1).
Z	Uncertainty range of +/- 6 inches. Selected based on the noding size of nuclear codes. No standard deviation (uniform uncertainty).
CHF Correlation	Standard deviation of 10.2% calculated from the BWCMV CHF test data base (Reference 4).
Thermal Hydraulic Code / Model	Uncertainty range of [ ] Value used in Reference 5. Standard deviation is calculated from the [ ] uncertainty value [ ].

TABLE 2

Comparison of the DNB Parameter Sensitivity of Different CHF Correlations With Consistent Axial Power Distributions

The following table shows the DNB sensitivity of each key parameter for the BWC CHF correlation (Oconee), the BWCMV CHF correlation (McGuire/Catawba), and the DCHF-1 CHF correlation (McGuire/Catawba). The first comparison is of a statepoint in the higher SDL area and the second is in the lower SDL area. The fluid and radial peaking conditions for each statepoint are given in the Appendices.

CHF Correlation	1.3 Peak @ 0.2 Z	1.3 Peak @ 0.8 Z
BWC	Statepoint 63	Statepoint 75
BWCMV	Statepoint 6	Statepoint 9
DCHF-1	Statepoint 6	Statepoint 9

1.3 Axial Peak, 0.2 Z

Parameter	BWC	BWCMV	DCHF-1
Power (%)			
Pressure (psi)			
Temperature (Deg F)			
Flow (%)			
FΔH (%)			
F_Z (%)			
Z (per 6 inches)			
SDL			

1.3 Axial Peak, 0.8 Z

Parameter	BWC	BWCMV	DCHF-1
Power (% RTP)			
Pressure (psi)			
Temperature (Deg F)			
Flow (%)			
FΔH (%)			
F_Z (%)			
Z (per 6 inches)			
SDL			

All values shown are in terms of % DNB per unit of parameter.

TABLE 3
SCD Transient Limiting Statepoints

The following table shows all the M/C transients currently evaluated with the SCD methodology. The determination of whether the transient uses the SCD approach is the value of all the key parameters (power, pressure, temperature, flow, peaking) at the point of MDNBR during the transient. All values listed are from the MDNBR point of the transient.

Transient	Core Power	Core Inlet Flow (Kgpm)	Core Inlet Temperature	Pressure	FΔH	F_Z	Z
Feed Line Break	[						
Partial Loss of RCS flow							
Total Loss of RCS Flow							
Uncontrolled RCCA Withdrawal / Subcritical							
*Uncontrolled RCCA Withdrawal / 100%							
*Uncontrolled RCCA Withdrawal / 100%							
Uncontrolled RCCA Withdrawal / 50%							
*Uncontrolled RCCA Withdrawal / 10%							
*Uncontrolled RCCA Withdrawal / 10%							
Single RCCA Withdrawal							
Statically Misaligned RCCA							
Dropped RCCA							]

* This accident was analyzed with two different reactivity insertion rates.

\# This accident was analyzed with a FΔH range of [

].

REFERENCES

1) Nuclear Design Methodology Using CASMO-3/SIMULATE-3P, DPC-NE-1004A, Duke Power Company, Charlotte, North Carolina, November 1992.

2) BWC Correlation for Critical Heat Flux, BAW-10143P-A, Babcock And Wilcox , Lynchburg, Virginia, April 1985.

3) Oconee Nuclear Station Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2003P-A, Duke Power Company, Charlotte, North Carolina, August 1988.

4) BWCMV Correlation Of Critical Heat Flux In Mixing Vane Grid Fuel Assemblies, BAW-10159P-A, Babcock And Wilcox, Lynchburg, Virginia, February, 1989.

5) McGuire and Catawba Nuclear Stations Core Thermal-Hydraulic Methodology Using VIPRE-01, DPC-NE-2004P-A, Duke Power Company, Charlotte, North Carolina, December 1991.

6) Thermal-Hydraulic Transient Analysis Methodology, DPC-NE-3000P-A, Revision 1, Duke Power Company, Charlotte, North Carolina, December 1991.

Revised Pages For Topical Report DPC-NE-2005P

The bar in the right hand margin notes revised lines.

TABLE 7. SDL Evaluation And Re-Submittal Criteria

The following table lists different events or conditions that would require an evaluation of the applicability of an approved SDL and the subsequent actions based on the results of the analysis.

CONDITION	ACTION
Revised uncertainty larger than the limiting value used in the original analysis.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
Revised uncertainty distribution.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
New statepoint.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
Minor modifications to the current fuel design.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
A modified CHF correlation.	SDL $\leq$ Limit, No Action. SDL > Limit, Specifically compensate.
Change to a new fuel design/fuel type.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
A new CHF correlation.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
Duke analysis of a non-Duke reactor.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.
New Thermal-Hydraulic Code.	Evaluate, submit a new Appendix for NRC approval regardless of SDL values.

FIGURE 2

REVISED SCD FLOWCHART

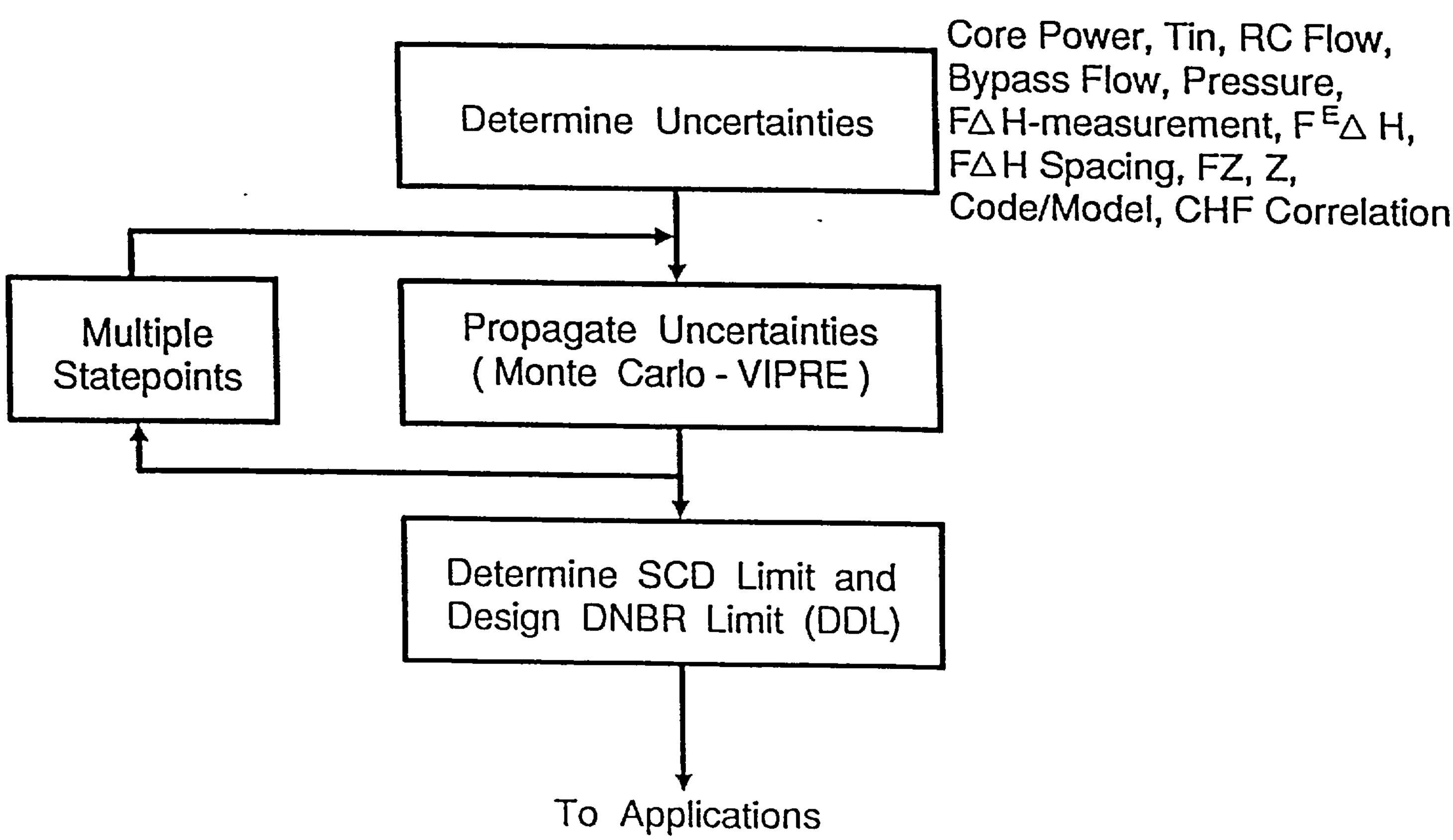

TABLE A-1. Oconee SCD Statepoints

TABLE A-4 Oconee Key Parameter Ranges

Parameter	Maximum	Minimum

All values listed in this table are based on the currently analyzed Statepoints. Ranges are subject to change based on future statepoint conditions.